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This calculation supersedes *Preclosure Consequence Analyses for License Application*, 000-00C-MGR0-00900-000-00A, ACC: ENG.20040701.0004.

Attachments	Total Number of Pages
A – Surface Facility Airborne Release Evaluation	6
B – Preclosure Radiation Dose Limits	4
C – MACCS2 Input and Output File Listing (CD attached)	2 pages and 1 CD

RECORD OF REVISIONS

9. No.	10. Reason For Revision	11. Total # of Pgs.	12. Last Pg. #	13. Originator (Print/Sign/Date)	14. Checker (Print/Sign/Date)	15. QER (Print/Sign/Date)	16. Approved/Accepted (Print/Sign)	17. Date
00A	Initial issue.	108	C-2	Sen-Sung Tsai	R.P. McDonnell J.A. Ziegler	Jerry Heaney	Dennis C. Richardson	
00B	Complete revision. Updated source terms and dose calculations for naval spent nuclear fuel and Hanford high-level radioactive waste. Deleted failed fuel oxidation from the Category 1 event sequence list. Included worker dose calculations for the Fuel Handling Facility and the spent nuclear fuel aging system. Updated references and revised to incorporate CR 3207 comments.	112 pages and 1 CD	C-2 of C-2	Sen-Sung Tsai SIGNATURE ON FILE 1/11/05	Kathryn L. Ashley SIGNATURE ON FILE 1/11/05	Charity Grano SIGNATURE ON FILE 1/12/05	Dennis C. Richardson SIGNATURE ON FILE	1/12/05

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TABLES

ACRONYMS

ALARA	as low as is reasonably achievable
ARF	airborne release fraction
BWR	boiling water reactor
CDE	committed dose equivalent
CEDE	committed effective dose equivalent
CHF	Canister Handling Facility
DCF	dose conversion factor
DDE	deep dose equivalent
DOE	U.S. Department of Energy
DPC	dual-purpose canister
DR	damage ratio; for example, fuel rod breakage fraction
DTF	Dry Transfer Facility (DTF 1 or DTF 2)
DTN	data tracking number
EDE	effective dose equivalent
FHF	Fuel Handling Facility
HEPA	high-efficiency particulate air [filter]
HLW	high-level radioactive waste
HVAC	heating, ventilation, and air-conditioning
INEEL	Idaho National Engineering and Environmental Laboratory
LDE	lens dose equivalent
LPF	leak path factor
MAR	material at risk
NRC	U.S. Nuclear Regulatory Commission

ACRONYMS (Continued)

PULF	release fraction pulverized into respirable sizes (< 10 μm) from a drop event
PWR	pressurized water reactor
RF	respirable fraction
SDE	shallow dose equivalent to skin
SFA	spent fuel assembly
SNF	spent nuclear fuel
SRS	Savannah River Site
SSCs	structures, systems, and components
TCRRF	Transportation Cask Receipt and Return Facility
TEDE	total effective dose equivalent
TODE	total organ dose equivalent

UNITS OF MEASURE

Ci	curie
cm	centimeter
cm^2	square centimeter
cm^3	cubic centimeter
ft	feet
g	gram
GWd	gigawatt days
hr	hour
in.	inch
kB	kilobyte
kW	kilowatt
m	meter
m^2	square meter
m^3	cubic meter
min	minute
mph	miles per hour
mrem	millirem
MTHM	metric tons of heavy metal
MTU	metric ton uranium
rem	roentgen equivalent man
s	second
Sv	Sieverts
μCi	microcurie
μm	micrometer
wt	weight

1. PURPOSE

Radiological consequence analyses are performed for potential releases from normal operations in surface and subsurface facilities and from Category 1 and Category 2 event sequences during the preclosure period. Surface releases from normal repository operations are primarily from radionuclides released from opening a transportation cask during dry transfer operations of spent nuclear fuel (SNF) in Dry Transfer Facility 1 (DTF 1), Dry Transfer Facility 2 (DTF 2), the Canister Handling facility (CHF), or the Fuel Handling Facility (FHF). Subsurface releases from normal repository operations are from resuspension of waste package surface contamination and neutron activation of ventilated air and silica dust from host rock in the emplacement drifts. The purpose of this calculation is to demonstrate that the preclosure performance objectives, specified in 10 CFR 63.111(a) and 10 CFR 63.111(b), have been met for the proposed design and operations in the geologic repository operations area. Preclosure performance objectives are discussed in Section 6.2.3 and are summarized in Tables 1 and 2.

Table 1. Performance Objectives for Normal Operations and Category 1 Event Sequences

Event Sequence Type	Dose Type	Performance Objectives		
		Worker[a]	Onsite Member of the Public	Offsite Member of the Public
Normal operations and Category 1	TEDE	5 rem/year[b]	100 mrem/year[c]	15 mrem/year[d] 100 mrem/year[b,c]
Normal operations and Category 1	Highest TODE	50 rem/year	NA	NA
Normal operations and Category 1	LDE	15 rem/year	NA	NA
Normal operations and Category 1	SDE	50 rem/year	NA	NA
Normal operations and Category 1	External dose: Highest of DDE, LDE, or SDE for the unrestricted area	NA	NA	2 mrem in any 1 hr[e]

NOTES: Radiation exposures and releases from Category 1 event sequences are aggregated per 10 CFR 63.111(b)(1).

[a]10 CFR 20.1201
[b]10 CFR 20.1101(b) provides for establishment for ALARA goals
[c]10 CFR 20.1301(a)(1)
[d]10 CFR 63.204
[e]10 CFR 20.1301(a)(2).

ALARA = as low as is reasonably achievable; DDE = deep dose equivalent; LDE = lens dose equivalent; mrem = one thousandth of a rem; NA = not applicable; rem = roentgen equivalent man; SDE = shallow dose equivalent to skin; TEDE = total effective dose equivalent; TODE = total organ dose equivalent.

Table 2. Performance Objectives for Category 2 Event Sequences

Event Sequence Type	Dose Type	Performance Objectives		
		Worker	Onsite Member of the Public	Offsite Member of the Public
Category 2	TEDE	NA	NA	5 rem/event
Category 2	Highest TODE	NA	NA	50 rem/event
Category 2	LDE	NA	NA	15 rem/event
Category 2	SDE	NA	NA	50 rem/event

LDE = lens dose equivalent; NA = not applicable; rem = roentgen equivalent man; SDE = shallow dose equivalent to skin; TEDE = total effective dose equivalent; TODE = total organ dose equivalent.

Source: 10 CFR 63.111(b)(2)

Category 1 event sequences are those natural events and human-induced event sequences that are expected to occur one or more times before permanent closure of the repository. Category 2 event sequences are other human-induced event sequences that have at least one chance in 10,000 of occurring before permanent closure of the repository. Event sequences that have less than one chance in 10,000 of occurring before permanent closure of the repository are designated as beyond Category 2 event sequences.

This calculation performs public dose calculations in Section 6.0. To support the license application, this calculation integrates the results from public dose calculations with worker dose results from the following calculations:

- *Canister Handling Facility Worker Dose Assessment* (BSC 2004a)
- *Category 1 Event Sequences Worker Dose Calculation* (BSC 2004b)
- *Dry Transfer Facility Worker Dose Assessment* (BSC 2004c)
- *Fuel Handling Facility Worker Dose Assessment* (BSC 2004d)
- *Geologic Repository Operations Area Worker Dose Assessment* (BSC 2004e)
- *GROA Airborne Release Dispersion Factor Calculation* (BSC 2004f)
- *Normal Operation Airborne Release Calculation* (BSC 2004g)
- *Remediation Facility Worker Dose Assessment* (BSC 2004h)
- *Subsurface Facility Worker Dose Assessment* (BSC 2004i)
- *Transportation Cask Receipt and Return Facility Worker Dose Assessment* (BSC 2004j).

The results of these worker dose calculations (BSC 2004a through 2004j) are used, as required by 10 CFR 63.111(a), to perform worker dose aggregation in Section 6.4.3. Aggregated doses are reported in Section 7.0.

Potential surface and subsurface releases from repository normal operations are discussed in Section 6.1.2. Event sequence frequency and categorization are discussed in Section 6.1.3. Listings of Category 1 and Category 2 event sequences and the types of waste forms and material at risk (MAR) involved in these event sequences are identified in Sections 6.1.4 and 6.1.5. This information is used as input to consequence analyses.

Results of public and worker dose calculations for normal operational releases and Category 1 event sequences, as well as bounding Category 2 event sequences, are summarized in Sections 6.3 and 6.4, which include:

- Total effective dose equivalent (TEDE)
- Committed dose equivalent (CDE) plus deep dose equivalent (DDE)
- Shallow dose equivalent to skin (SDE)
- Lens dose equivalent (LDE).

Direct radiation exposures during normal operations and Category 1 and Category 2 event sequences are discussed in Section 6.3.5. Atmospheric dispersion factor (χ/Q) values for surface facility releases are presented in Attachment A.

2. QUALITY ASSURANCE

This analysis is subject to *Quality Assurance Requirements and Description* (DOE 2004a, Section 2.2.2), because it pertains to preclosure safety analysis. This analysis was performed in accordance with AP-3.12Q, *Design Calculations and Analyses*, and is subject to the requirements of AP-3.13Q, *Design Control.* Unverified design inputs are identified and tracked in accordance with AP-3.15Q, *Managing Technical Product Inputs*.

3. METHOD

Potential releases from normal operations in surface and subsurface facilities are discussed in Section 6.1.2. The methodology used for evaluating radiological consequences from normal operations and Category 1 event sequences is presented in Section 3.1. The methodology used for evaluating radiological consequences from Category 2 event sequences is presented in Section 3.2. Potential Category 1 and Category 2 event sequences and the types of waste forms, and quantities of the waste forms, involved in the event sequences are identified in *Categorization of Event Sequences for License Application* (BSC 2004k). The categorization analysis identifies Category 1 and Category 2 event sequences.

Calculations are performed in this document using the MACCS2 V.1.12 computer code (MELCOR Accident Consequence Code System for the Calculation of the Health and Economic Consequences of Accidental Atmospheric Radiological Releases) (ORNL 1998). MACCS2 (ORNL 1998) calculates doses to onsite and offsite members of the public resulting from surface and subsurface releases during normal operations and Category 1 and Category 2 event sequences. MACCS2 (ORNL 1998) considers potential exposure pathways, including inhalation, resuspension inhalation, ingestion, air submersion, and groundshine. A description of the MACCS2 (ORNL 1998) computer program is given in Section 5.1.

Four dose measures, as follows, are applicable to normal operations and to Category 1 and Category 2 event sequences:

- **TEDE**–For purposes of assessing doses to workers, the TEDE is equal to the sum of the DDE for external exposures and the committed effective dose equivalent (CEDE) for internal exposures (10 CFR 63.2). For purposes of assessing doses to members of the public, the TEDE is equal to the sum of the effective dose equivalent (EDE) for external exposures and the CEDE (10 CFR 63.2) for internal exposures. The CEDE is calculated using the effective inhalation dose conversion factor (DCF). The EDE is calculated using the effective air submersion DCF. The TEDE includes ingestion and groundshine doses in addition to inhalation and submersion doses. In assessing compliance with the individual radiation protection standard, the DDE is replaced by the EDE as recommended in U.S. Nuclear Regulatory Commission (NRC) guidance on the use of the DDE and EDE for external exposure (66 FR 55732, p. 55735).

- **The Highest of the Total Organ Dose Equivalent (TODE)**–The highest TODE is equal to the sum of the highest CDE and the DDE. The organs evaluated to determine the highest CDE are the lungs, breasts, gonads, red marrow, bone surface, thyroid, and remainder. The remainder is not an organ, but rather a weighted combination of five remaining organs or tissues receiving the highest doses, such as the liver, kidneys, spleen, brain, small intestine, upper large intestine, lower large intestine, or other organs, excluding the skin, lens of the eye, and extremities, as stated in Federal Guidance Report No. 11 (Eckerman et al. 1988, p. 6). The DDE, which is added to the highest CDE, is equal to that used to calculate the TEDE. In assessing compliance with the individual radiation protection standard, the DDE is replaced by the EDE as recommended in NRC guidance on the use of the DDE and EDE for external exposure (66 FR 55732, p. 55735).

- **Lens Dose Equivalent**–In Federal Guidance Report No. 11 (Eckerman et al. 1988, Table 2.1), only one lens of the eye dose DCF of ^{83m}Kr can be found. The LDEs are not calculated using the lens of the eye DCFs, because the lens of the eye DCFs given in Federal Guidance Report No. 11 are incomplete. NUREG-1567 (NRC 2000, p. 9-14) states that compliance with the lens of the eye dose limit is achieved if the sum of the SDE and the TEDE does not exceed 15 rem.

- **Shallow Dose Equivalent to Skin**–The dose to the skin is from the air submersion and groundshine pathways. SDEs are calculated using MACCS2 (ORNL 1998), which is based on the DCFs for air submersion and for exposure to contaminated ground surfaces in Federal Guidance Report No. 12 (Eckerman and Ryman 1993, Tables III.1 and III.3).

3.1 METHODOLOGY FOR NORMAL OPERATIONS AND CATEGORY 1 EVENT SEQUENCES

This section discusses the methodology used for evaluating radiological consequences from normal operations in the surface and subsurface facilities and Category 1 event sequences.

3.1.1 Normal Operations and Category 1 Event Sequence Performance Objectives

Performance objectives for normal operations and Category 1 event sequences are provided in 10 CFR Part 20 and 10 CFR Part 63, and are summarized in Table 1. They are discussed in Section 6.2.3.

3.1.2 Normal Operations and Category 1 Event Sequence Public Dose Methodology

This section discusses the method for calculating doses from normal operations and Category 1 event sequences and dose aggregation as required by 10 CFR 63.111(a)(2).

In releases from normal operations, radioactive materials are conservatively assumed to be released inside a waste transfer cell when a transportation cask is opened. It is assumed that the heating, ventilation, and air-conditioning (HVAC) system is operating and the radioactive material is vented through the exhaust stack and is released as a radioactive plume that is dispersed en route to the site boundary. This results in an acute individual exposure during plume passage and a chronic individual exposure to ground contamination and contaminated food for 1 year after plume passage. A release duration of 24 hr is assumed for surface and subsurface releases (Section 4, Assumption 4.6).

MACCS2 (ORNL 1998) is designed to simulate accidents with a release duration of 24 hr or less. MACCS2 (ORNL 1998), however, is used to simulate normal operations for which the release duration is one year. To model normal operations using MACCS2 (ORNL 1998), a release duration of 24 hr instead of one year is used. The use of a 24-hr release duration is conservative because: (1) a 24-hr release duration has a shorter time for radionuclide decay than a 1-year release duration, and (2) a 24-hr release duration results in a longer duration at higher ground concentrations of radionuclides and, therefore, higher resuspension inhalation, groundshine, and ingestion doses. The inhalation dose is not impacted by the use of a shorter release duration because it is not a function of the release duration. This is confirmed by sensitivity analysis (Section 6.3.4.1), which shows that modeling normal operations with a 24-hr duration instead of a 1-year duration is conservative because the shorter duration MACCS2 (ORNL 1998) run shows higher doses (Section 6.3.4.1).

No credit is taken for plume meandering or for plume rise from initial thermal energy to ensure that the χ/Q value is conservative at the site boundary. Credit is taken for high-efficiency particulate air (HEPA) filters to remove airborne radionuclides. For normal operations, credit is taken for atmospheric dispersion of radionuclides released via the exhaust stack (Section 4, Assumption 4.7). Therefore, the release is modeled as an elevated release with no plume rise and the building wake effect on the plume is considered.

For Category 1 event sequences involving drops or collisions, or both, of SNF, it is assumed that radioactive materials are released inside a waste transfer cell. It is assumed that the HVAC system is operating and that radioactive material is vented through the exhaust stack as a radioactive plume. The plume is dispersed en route to the site boundary, resulting in an acute individual exposure during plume passage and in a chronic individual exposure to ground contamination and contaminated food for 1 year after plume passage (Section 4, Assumption 4.5). To ensure that the χ/Q value is conservative at the site boundary, no credit is taken for plume meandering or for plume rise from initial thermal energy. Credit is taken for HEPA filters to remove airborne radionuclides. For Category 1 event sequences, no credit is taken for the stack height and the release is modeled as a ground-level release (Section 4, Assumption 4.17), and the building wake effect is considered. A release duration of 1 hr is assumed (Section 4, Assumption 4.5).

The χ/Q values are calculated internally in MACCS2 (ORNL 1998) using hourly meteorological data gathered at the Yucca Mountain site from 1998 through 2002 (Section 6.2.1.1.2). MACCS2 (ORNL 1998) uses 8,760 sets of hourly data on wind speed, wind direction, atmospheric stability class, and rain intensity collected over a period of 1 year to calculate χ/Q values. This process generates 8,760 χ/Q values, which includes the 16 directional sectors, wind speeds, and atmospheric stability classes collected in the annual data for 1999. The χ/Q value output from MACCS2 (ORNL 1998) is a mean, 50th percentile, 95th percentile, or 99.5th percentile evaluated at a 11 km distance from the release point for a surface release, or at a 8 km distance for a subsurface release (Section 6.2.1.5 and Attachment B, Figure B-1).

The χ/Q values are determined regardless of wind direction or sector for the distance of interest. The χ/Q values are overall site χ/Q values and are not tied to any particular sector, such as the sector to the west site boundary. Because all sector meteorological data are used in χ/Q calculations and the receptor location (i.e., 11 km from a DTF to the west site boundary) is the shortest distance from the release point, the resultant χ/Q values are the most conservative and bound all site boundary locations.

The sector-independent MACCS2 (ORNL 1998)-calculated mean χ/Q value of 2.69E-06 s/m^3 (Attachment A, Table A-1) is at least one order of magnitude higher than the maximum sector chronic χ/Q value of 2.0E-07 s/m^3 at a distance of 11 km, using Regulatory Guide 1.111, Equation 3, as calculated in *Calculations of Acute and Chronic "Chi/Q" Dispersion Estimates for a Surface Release* (CRWMS M&O 1999a, Table 6c).

Category 1 event sequences and normal operational releases use the mean χ/Q value. The mean χ/Q value was found to be more conservative than the 50th percentile χ/Q value.

Source terms for spent fuel assemblies (SFAs) from pressurized water reactors (PWRs) and boiling water reactors (BWRs), with four different combinations of initial enrichment, burnup, and decay time, are presented in Table 3.

Table 3. Average and Maximum Boiling Water Reactor and Pressurized Water Reactor Spent Fuel Assemblies

Assembly	Initial Enrichment (Percent)	Burnup (GWd/MTU)	Decay Time (Years)
Average PWR[a]	4.0	48	25
Maximum PWR[a]	5.0	80	5
Average BWR[b]	3.5	40	25
Maximum BWR[b]	5.0	75	5

NOTES: [a] BSC 2004l , p. 27
[b] BSC 2003a , p. 46.

BWR = boiling water reactor; GWd = gigawatt days; MTU = metric ton uranium; PWR = pressurized water reactor; SFAs = spent fuel assemblies.

Radionuclide inventories in curies per fuel assembly (Ci/FA) for the Average and Maximum PWR and the Average and Maximum BWR SFAs are presented in Table 4. Both the Average PWR and the Average BWR SFAs are used to calculate the mean doses and the 50[th] percentile doses for Category 1 event sequences. The calculated mean dose is used because it is higher than the calculated 50[th] percentile dose.

Table 4. Boiling Water Reactor and Pressurized Water Reactor Radionuclide Inventories

Radionuclide	Average PWR (Ci/FA)[a]	Maximum PWR (Ci/FA)[a]	Average BWR (Ci/FA)[b]	Maximum BWR (Ci/FA)[b]
Ac-227	1.61E-05	0.00E+00	0.00E+00	0.00E+00
Am-241	1.98E+03	8.79E+02	5.58E+02	2.66E+02
Am-242	6.36E+00	1.01E+01	2.16E+00	3.39E+00
Am-242m	6.39E+00	1.02E+01	2.17E+00	3.41E+00
Am-243	2.20E+01	6.00E+01	5.34E+00	1.93E+01
Ba-137m	3.88E+04	9.89E+04	1.31E+04	3.65E+04
C-14	3.32E-01	5.35E-01	1.76E-01	3.16E-01
Cd-113m	7.66E+00	4.31E+01	2.26E+00	1.39E+01
Cl-36	6.80E-03	1.05E-02	2.93E-03	4.99E-03
Cm-242	5.26E+00	3.56E+01	1.78E+00	1.13E+01
Cm-243	1.03E+01	4.19E+01	2.47E+00	1.12E+01
Cm-244	1.36E+03	1.40E+04	2.55E+02	3.94E+03
Cm-245	3.07E-01	1.79E+00	4.03E-02	3.53E-01
Cm-246	1.04E-01	1.21E+00	1.45E-02	2.96E-01
Co-60	3.13E+02	5.99E+03	4.39E+01	8.53E+02
Cs-134	2.52E+01	4.05E+04	6.32E+00	1.15E+04
Cs-135	3.50E-01	6.34E-01	1.39E-01	2.82E-01
Cs-137	4.11E+04	1.05E+05	1.39E+04	3.87E+04
Eu-154	6.71E+02	6.15E+03	1.75E+02	1.80E+03
Eu-155	5.15E+01	1.80E+03	1.60E+01	6.28E+02
Fe-55	3.46E+00	7.27E+02	1.09E+00	2.34E+02
H-3	1.14E+02	4.95E+02	3.95E+01	1.76E+02
I-129	2.19E-02	3.60E-02	7.42E-03	1.35E-02

Table 4. Boiling Water Reactor and Pressurized Water Reactor Radionuclide Inventories (Continued)

Radionuclide	Average PWR (Ci/FA)[a]	Maximum PWR (Ci/FA)[a]	Average BWR (Ci/FA)[b]	Maximum BWR (Ci/FA)[b]
Kr-85	1.13E+03	5.79E+03	3.81E+02	2.03E+03
Nb-93m	1.30E+01	4.87E+01	4.73E-01	1.22E+00
Nb-94	8.39E-01	1.37E+00	1.87E-02	3.38E-02
Ni-59	2.09E+00	2.96E+00	5.02E-01	7.78E-01
Ni-63	2.52E+02	4.52E+02	5.86E+01	1.16E+02
Np-237	2.47E-01	4.01E-01	6.89E-02	1.33E-01
Pa-231	2.97E-05	4.18E-05	1.39E-05	2.94E-05
Pd-107	8.41E-02	1.60E-01	2.65E-02	5.69E-02
Pm-147	1.19E+02	2.29E+04	3.98E+01	7.46E+03
Pu-238	2.29E+03	6.80E+03	5.85E+02	2.11E+03
Pu-239	1.77E+02	1.83E+02	5.35E+01	5.36E+01
Pu-240	3.18E+02	4.01E+02	1.14E+02	1.48E+02
Pu-241	2.46E+04	8.00E+04	6.78E+03	2.25E+04
Pu-242	1.64E+00	3.34E+00	5.08E-01	1.26E+00
Ru-106	1.23E-02	1.33E+04	3.00E-03	3.29E+03
Sb-125	9.71E+00	2.14E+03	2.89E+00	6.21E+02
Se-79	4.57E-02	7.35E-02	1.59E-02	2.88E-02
Sm-151	2.11E+02	3.19E+02	5.39E+01	8.22E+01
Sn-126	3.85E-01	6.83E-01	1.27E-01	2.52E-01
Sr-90	2.72E+04	6.52E+04	9.54E+03	2.52E+04
Tc-99	8.98E+00	1.34E+01	3.20E+00	5.35E+00
Th-230	1.48E-04	3.33E-05	6.09E-05	2.05E-05
U-232	2.04E-02	5.97E-02	4.63E-03	2.00E-02
U-233	3.79E-05	2.42E-05	1.06E-05	0.00E+00
U-234	6.77E-01	5.21E-01	2.50E-01	2.26E-01
U-235	7.37E-03	3.28E-03	2.62E-03	9.43E-04
U-236	1.72E-01	2.23E-01	6.26E-02	9.55E-02
U-238	1.48E-01	1.42E-01	6.32E-02	6.07E-02
Y-90	2.72E+04	6.53E+04	9.54E+03	2.52E+04
Zr-93	8.94E-01	1.41E+00	3.39E-01	6.03E-01

NOTES: [a] BSC 2004l , Attachment IX
[b] BSC 2003a, Attachment XIII.

BWR = boiling water reactor; Ci/FA = curies per fuel assembly; PWR = pressurized water reactor; SNF = spent nuclear fuel.

Airborne release fractions (ARFs) and respirable fractions (RFs) for intact and damaged commercial SNF and naval SNF are shown in Table 5. Fuel rods with hairline cracks and pinhole leaks are treated as intact fuel rods as stated in ISG-1 (NRC 2002). ARFs and RFs for intact commercial SNF assemblies (Table 5) are used for normal operations and Category 1 event sequences. ARFs and RFs for intact single commercial SNF rods (Table 5) are used for Category 2 event sequences because drops or collisions of canistered intact commercial SNF rods into transportation casks and waste packages are assumed to be Category 2 event sequences (Section 4, Assumption 4.9).

Table 5. Spent Nuclear Fuel Airborne Release Fractions and Respirable Fractions

Nuclide	Airborne Release Fraction (ARF)[a]	Respirable Fraction (RF)[a]	Effective Release Fraction (ARF × RF)[a]
Breach of Intact SNF Assemblies and Rods [b]			
H-3	0.3[c]	1.0[c]	0.3[c]
Kr-85	0.3[c]	1.0[c]	0.3[c]
I-129	0.3[c]	1.0[c]	0.3[c]
Cs	2.0E-04[c]	1.0[c]	2.0E-04[c]
Sr[d]	3.0E-05[e]/5.9E-07[f]	5.0E-03[e]/1.0[f]	1.5E-07[e]/5.9E-07[f]
Ru	2.0E-04[c]	1.0[c]	2.0E-04[c]
Crud	1.5E-02[c, g]	1.0[c]	1.5E-02[c]
Fuel Fines other than Sr	3.0E-05[e]/5.9E-07[f]	5.0E-03[e]/1.0[f]	1.5E-07[e]/5.9E-07[f]
Oxidation of Damaged Commercial SNF in Air[h]			
H-3	0.3	1.0	0.3
Kr-85	0.3	1.0	0.3
I-129	0.3	1.0	0.3
Cs	2.0E-04	1.0	2.0E-04
Sr[d]	1.2E-04	1.0	1.2E-04
Ru	2.0E-04	1.0	2.0E-04
Crud	1.5E-02[g]	1.0	1.5E-02
Fuel Fines other than Sr	1.2E-04	1.0	1.2E-04

NOTES: [a] ARFs and RFs are for intact assemblies and intact single fuel rods; fuel rods with hairline cracks and pinhole leaks are treated as intact fuel rods
[b] Source: BSC 2004m, Section 6.2.1.3 and Table 11
[c] Intact commercial SNF assemblies and rods and naval SNF
[d] Sr is treated as fuel fines
[e] Intact commercial SNF assemblies and naval SNF
[f] Intact commercial SNF rods
[g] For crud, the value shown is the "effective ARF," which is the product of a crud spallation fraction of 0.15 and an ARF of 0.1 (BSC 2004m, Section 6.2.1.3)
[h] Section 4, Assumption 4.16, and Section 6.2.1.3.

ARF = airborne release fraction; RF = respirable fraction; SNF = spent nuclear fuel.

Damaged commercial SNF could be oxidized in air during fuel transfer operations inside a waste transfer cell. In Table 5, for oxidation of damaged commercial SNF in air, the ARFs and RFs for fission product gases, volatile species, and crud are based on Assumption 4.16 (Section 4) and the ARFs and RFs for fuel fines are developed in Section 6.2.1.3.

Potential radiation doses to members of the public that come from inhalation, resuspension inhalation, ingestion, air submersion, and groundshine pathways are considered for normal operations and for Category 1 event sequences. Whole body and organ doses to a member of the public at the site boundary by inhalation, resuspension inhalation, ingestion, air submersion, and groundshine pathways are calculated using MACCS2 (ORNL 1998, Section 5).

Dose Aggregation–Total annual public dose is based on contributions from four sources:

- Category 1 event sequences
- Normal operational releases from surface facilities
- Normal operational releases from the subsurface repository
- Direct radiation dose from contained radiation sources.

The direct radiation dose to members of the public is expected to be very low because radiation sources, such as transportation casks, waste packages, and site specific casks are processed at locations far away from the site boundary. Therefore, only the Category 1 annual dose and normal operations annual doses are considered for the dose aggregation calculation as described by:

$$D_{TOT} = D_{Cat.1} + D_{NO}^{SF} + D_{NO}^{Sub} \qquad \text{(Eq. 1)}$$

where,

$D_{Cat.1}$	=	Annual dose from Category 1 event releases (mrem/year)
D_{NO}^{SF}	=	Annual dose from normal operational releases from surface facilities (mrem/year)
D_{NO}^{Sub}	=	Annual dose from normal operational releases from the subsurface repository (mrem/year).

$D_{Cat.1}$, D_{NO}^{SF}, and D_{NO}^{Sub} are calculated using MACCS2 (ORNL 1998). Radiological release estimates in Ci/year for each of these three components are used as input to the MACCS2 (ORNL 1998) dose calculations. Annual releases from normal operations in surface facilities and normal operations in the subsurface repository are summarized in Section 6.1.2. Annual dose from a Category 1 event sequence is calculated using:

$$D_{Cat.1} = \sum_{i=1}^{n} D_i \cdot f_i \qquad \text{(Eq. 2)}$$

where,

i	=	Index for a given Category 1 event sequence; i = 1, 2,...n
n	=	Total number of Category 1 event sequences
D_i	=	Annual dose from event sequence i (rem/event)
f_i	=	Frequency of event sequence i (events/year).

To show compliance with 10 CFR 63.111(a)(2), the calculated annualized dose, D_{TOT}, is compared with the regulatory dose limit of 15 mrem/year at the site boundary. In addition, the calculated TEDE from each Category 1 event sequence and any combination of Category 1 event sequences that can occur in 1 year are also compared with the regulatory dose limit of 15 mrem/year at the site boundary.

3.1.3 Worker Dose Methodology

As stated in Section 1, the results from worker dose calculations (BSC 2004a through 2004j) are reported in this calculation and used in Section 6.4.3 to perform worker dose aggregation. This section provides a more detailed discussion on the methodologies used in the worker dose calculations that are not provided in the referenced calculations (BSC 2004a through 2004j).

This section describes the methods used to calculate worker dose during normal operations and Category 1 event sequences. Regulation 10 CFR 63.111(a)(1) requires that the geologic repository operations area must meet the requirements of 10 CFR Part 20 for radiation protection. Shielding must take normal operations and Category 1 event sequences into consideration. The primary objective of shielding is to maintain occupational radiation exposures as low as is reasonably achievable (ALARA), and within the exposure dose limits specified in 10 CFR 20.1201. The controlling dose limit for workers (Table 1) is a TEDE of 5 rem/year in 10 CFR 20.1201(a)(1). The general approach used to estimate worker doses is to estimate radiation levels in occupied areas, to determine the personnel requirements and duration of activities in these areas, and to combine the dose rates, times, and personnel estimates to generate the worker dose estimates.

ALARA design goals, as described in *Project Design Criteria Document* (BSC 2004n, Section 4.9.3.3), for occupational workers ensure that both individual and collective annual doses are maintained at ALARA levels during normal operations and as a result of Category 1 event sequences. Category 1 event sequences are included in ALARA dose assessments.

Sections 3.1.3.1 and 3.1.3.2 present the methodologies for calculating worker dose and the potential dose consequences from surface and subsurface releases during normal operations. Among the surface facilities, only DTF 1, DTF 2, and the FHF have a potential for airborne releases of radionuclides during normal operations or following a Category 1 event sequence. Section 3.1.3.5 presents the methodology for calculating the worker dose from direct radiation.

3.1.3.1 Worker Dose From Airborne Releases

It is assumed that, for radionuclides released from a waste transfer cell within a surface facility, the HVAC system is operating and the airborne radionuclides are vented through the building exhaust stack, then disperse into the atmosphere and reenter the building through the building ventilation system air intakes (Section 4, Assumption 4.14). For radionuclides released from the subsurface facility, it is assumed that the airborne radionuclides are dispersed into the atmosphere and reenter the subsurface facility through the subsurface ventilation system air intakes (Section 4, Assumption 4.14).

The TEDE dose measure, described in ISG-5 (NRC 2003a, p. 10), is expressed as:

$$TEDE = CEDE + EDE = \sum_j D^{inh}_{j,effective} + \sum_j D^{ing}_{j,effective} + \sum_j D^{ext}_{j,effective} \qquad \text{(Eq. 3)}$$

where,

$TEDE$	=	Total effective dose equivalent (rem)
$CEDE$	=	Committed effective dose equivalent (rem)
EDE	=	Effective dose equivalent (rem)
$D^{inh}_{j,effective}$	=	Whole body effective inhalation dose from the j^{th} nuclide (rem)
$D^{ing}_{j,effective}$	=	Whole body effective ingestion dose from the j^{th} nuclide (rem)
$D^{ext}_{j,effective}$	=	Whole body effective external dose from the j^{th} nuclide (rem)

The TODE (= CDE + DDE) measure is expressed as (NRC 2003a, p. 10):

$$TODE_k = CDE_k + EDE_k = \sum_j D^{inh}_{j,k} + \sum_j D^{ing}_{j,k} + \sum_j D^{ext}_{j,k} \quad \text{where k} \neq \text{effective or skin} \qquad \text{(Eq. 4)}$$

where,

CDE_k	=	Committed dose equivalent to the k^{th} organ (rem)
$D^{inh}_{j,k}$	=	Radiation dose from the j^{th} nuclide to the k^{th} organ from inhalation (rem)
$D^{ing}_{j,k}$	=	Radiation dose from the j^{th} nuclide to the k^{th} organ from ingestion (rem)
$D^{ext}_{j,k}$	=	Radiation dose from the j^{th} nuclide to the k^{th} organ from external exposure (rem)
k	=	Organ index, where organs are gonads, breast, lungs, red marrow, bone surface, thyroid, and remainder.

The external dose is the sum of the groundshine dose and the air submersion dose. The groundshine dose has been shown (Section 6.3) to be a small contribution to the total dose and therefore the external dose is approximated by the air submersion dose. For example, MACCS2 (ORNL 1998) Run 4 (Section 6.3) shows that the groundshine dose at 100 m is about 5.5E-04 mrem, which is less than 0.1 percent of the inhalation dose.

The ingestion dose is not calculated for onsite workers because no ingestion of contaminated food, water, or soil is expected.

For airborne releases of radionuclides through building vents, the previous inhalation and external doses can be further expressed as (NRC 2003a, pp. 9 and 10):

$$D_{j,k}^{inh} = ST_j \times FA \times \frac{\chi}{Q} \times BR \times conv \times DCF_{j,k}^{inh} \qquad \text{(Eq. 5)}$$

$$D_{j,k}^{sub} = ST_j \times FA \times \frac{\chi}{Q} \times conv \times DCF_{j,k}^{sub} \qquad \text{(Eq. 6)}$$

where,

ST_j	=	Release source term per fuel assembly for the j^{th} nuclide (Ci/FA)
FA	=	Number of SFAs involved in the release per year or per event; FA = 6,316 PWR SFAs for normal operations per year and $FA = 2$ PWR SFAs per event for Category 1 event sequences
$\frac{\chi}{Q}$	=	Atmospheric dispersion factor (s/m^3)
BR	=	Breathing rate = 3.33×10^{-4} m^3/s (NRC 2000, p. 9-13)
$DCF_{j,k}^{inh}$	=	Inhalation DCF of the j^{th} nuclide for the k^{th} organ in Sv/Bq (Eckerman et al. 1988, Table 2.1)
$DCF_{j,k}^{sub}$	=	Air submersion DCF of the j^{th} nuclide for the k^{th} organ [(Sv-m^3)/(Bq-s)] (Eckerman and Ryman 1993, Table III.1)
$conv$	=	DCF unit conversion factor: 3.7×10^{12} (rem-Bq)/(Ci-Sv) (Eckerman and Ryman 1993).

The nominal annual capacity and rate of receipt is 3,000 MTHM/year, as presented in *Civilian Radioactive Waste Management System Requirements Document* (DOE 2004b, Table 1), or about 6,316 PWR SFAs per year based on 0.475 MTU/SFA. It is assumed that 1 percent (Section 4, Assumption 4.3) of fuel rods in these SFAs are damaged and that radionuclides are released to the environment. It is assumed that for Category 1 event sequences, an SFA is dropped back into a transportation cask and damages another SFA inside the cask during the dry transfer of SNF from a transportation cask to a waste package.

Annual average χ/Q values are generated using the NRC-sponsored computer code, *Software Code: ARCON.* V.96 (Atmospheric Relative CONcentrations in Building Wakes [ARCON96]), (BSC 2003b), as described in NUREG/CR-6331 (Ramsdell and Simonen 1997). ARCON96 (BSC 2003b) was developed by Pacific Northwest National Laboratory for the NRC to calculate χ/Q values in plumes for nuclear power plants at control room air intakes in the vicinity of the release point. ARCON96 (BSC 2003b) implements a straight-line Gaussian dispersion model with dispersion coefficients modified to account for low wind meander and building wake effects. ARCON96 (BSC 2003b) also accounts for variations in the location of release points.

The χ/Q value from normal operational releases and Category 1 event sequences are predicted using ARCON96 (BSC 2003b). For each receptor of interest, a cumulative probability distribution of χ/Q is constructed by the code for release time periods of 2 hr, 4 hr, 8 hr, and 12 hr. The 8,760-hr probabilistic χ/Q distribution is used to determine the annual average and the 50 percent weather probability median χ/Q values for the receptor of interest. The larger of the annual average or median χ/Q value is used to calculate the annual average dose to workers from airborne radionuclides during normal operations and from Category 1 event sequences.

The annual average χ/Q value represents the average dilution of an airborne contamination from atmospheric mixing and turbulence based on the site-specific atmospheric conditions, the relative configuration of the release point and the receptor, and the distance from the release point to the receptor of interest. It is the ratio of the average contaminant air concentration at the receptor to the contaminant release rate at the release point, and it is used to determine the dose consequences for a receptor based on the quantity released, the atmospheric conditions, and the distance to the receptor.

The χ/Q values at various onsite receptor locations for airborne releases from surface and subsurface facilities were calculated using ARCON96 (BSC 2003b) and the results of the calculations are reported in BSC (2004f, Section 6.3). The maximum χ/Q values at receptors for each of the seven release sources were used to calculate the maximum worker doses for releases from the surface and subsurface facilities from normal operations and Category 1 event sequences, respectively. These χ/Q values represent the dispersion factors estimated at the air intake point of the surface and subsurface facilities.

The airborne release source term is the amount of radioactive material in Ci that is released to the air per assembly under normal operating conditions or following a Category 1 event sequence. The airborne release source term, ST_j, is calculated by the following, from DOE-HDBK-3010-94 (DOE 1994, p. 1-2):

$$\text{Release Source Term (Ci/FA) } ST_j = MAR_j \times DR \times ARF \times LPF \qquad \text{(Eq. 7)}$$

where,

MAR_j	=	Material at risk; commercial SNF radionuclide inventory per fuel assembly (Ci/FA)
DR	=	Damage ratio; rod breakage fraction (DR = 1.0)
ARF	=	Airborne release fraction; the fraction of processed commercial SNF that is suspended or released from the SFAs
LPF	=	Leak path factor; the fraction of airborne released material that discharges into the atmosphere from the HVAC exhaust systems; LPF = 1.0E-04 for normal operations and Category 1 event sequences.

The SDE is equal to (NRC 2003a, p. 10):

$$SDE = \sum_{j} D_{j,skin}^{sub} \qquad \text{(Eq. 8)}$$

where,

SDE = Shallow dose equivalent to skin (rem)

$D_{j,skin}^{sub}$ = Radiation dose from the j^{th} nuclide to the skin from air submersion (rem).

It is stated in NUREG-1567 (NRC 2000, p. 9-14), that compliance with the LDE limit is achieved if the sum of the SDE and the TEDE does not exceed 15 rem; the LDE may be expressed as:

$$LDE = TEDE + SDE \qquad \text{(Eq. 9)}$$

where,

LDE = Lens dose equivalent (rem).

The assumptions in the following paragraph are consistent with the release fractions assumed in NUREG-1567 (NRC 2000, p. 9-12) and ISG-5 (NRC 2003a, Attachment, Table 7.1).

For normal operations, 1 percent (Section 4, Assumption 4.3) of the fuel rods received are assumed to have cladding breaches that could cause the release of fission product gases, volatile species, and particulates in the gap region. It is assumed that for Category 1 event sequences involving drops and collisions, 100 percent of the fuel rods have failed because of the event (Section 4, Assumption 4.10). Of the total SFA radioactive inventory, the ARFs given in Table 5 are used for Category 1 event sequences involving drops and collisions.

Strontium isotopes including ^{90}Sr are considered as nonvolatile materials and are, therefore, treated as fuel particles (BSC 2004m, Section 6.2.1.2 and Table 11).

For crud, a crud spallation fraction of 0.15 (BSC 2004m, Section 6.2.1.3) is used, which means that 15 percent of the crud becomes loose from the fuel rod surfaces under normal conditions and Category 1 event sequences. Of the loose crud, 10 percent (BSC 2004m, Section 6.2.1.3) becomes airborne and released during normal operations and Category 1 event sequences. An ARF of 10 percent represents the bounding release fraction for the case when venting gases pressurize the volume in which there exists loose powdering surface contamination (DOE 1994, p. 5-22). An ARF of 10 percent also bounds ARFs for other potential release mechanisms, such as venting of pressurized powders or pressurized gases through a powder (DOE 1994, Section 4.4.2.3) and suspension of surface contamination from solid material by impact and vibration shock (DOE 1994, pp. 5-7 and 5-24). The effective crud ARF is defined as the product of the crud spallation fraction (0.15) and the crud ARF (0.1). An effective crud ARF of 0.015 is given in Table 5.

For calculating inhalation dose, Equation 7 is multiplied by the RFs to calculate respirable release source terms. The RF is the respirable fraction of airborne radionuclide particles having an aerodynamic equivalent diameter of 10 μm or less.

For estimating respirable release source terms for oxidation event sequences, an RF of 1.0 is used (Table 5). For estimating respirable release source terms for Category 1 event sequences involving drops and collisions, the following RFs are used (Table 5):

- RF of 1 for fission product gases and volatile nuclides
- RF of 0.005 for fuel fines
- RF of 1 for crud.

In Equation 7, the release source term per PWR or BWR assembly for radionuclide *j*, ST_j, is calculated by multiplying the Average PWR or the Average BWR radionuclide inventory per assembly (MAR_j) for that radionuclide (Table 4) by the damage ratio (DR), the ARF (Table 5), and the leak path factor (LPF).

A 25-year ^{60}Co activity of 2.35 Ci (Section 6.2.1.2) per PWR assembly in crud on the surface of commercial SNF assemblies is added to the release source term. The use of 25-year old crud is consistent with the use of the Average PWR or the Average BWR radionuclide inventory, which has a cooling time of 25 years. Again, the crud release source term is calculated by multiplying the ^{60}Co activity of 2.35 Ci/FA by the ARF for intact commercial SNF assemblies (Table 5).

The source term includes a list of radionuclides selected from the output of a point depletion of and decay calculation using a computer code, such as the SAS2H sequence in SCALE V4.3, from NUREG/CR-0200 (ORNL 1997). Source term radionuclides are based on selection criteria in NUREG-1567 (NRC 2000, p. 9-11) and ISG-5 (NRC 2003a, Attachment, Section 3). For confinement analysis, the source term, as a minimum, includes activity from ^{60}Co in crud, activity from iodine, other fission products that contribute greater than 0.1 percent of design basis fuel activity, and actinide activity that contributes greater than 0.01 percent of the design basis activity. Individual radionuclide activity and its contribution in percent for PWR and BWR source term nuclides are presented in BSC (2004b, Tables I-1 to I-3). The nuclides ^{14}C and ^{36}Cl are included in the tables (BSC 2004b, Tables I-1 to I-3) because of a potential release into the atmosphere as gaseous nuclides.

For normal operations and Category 1 event sequences, a two-stage HEPA filtration system with a particulate removal efficiency of 99 percent per stage with a HEPA LPF of 0.01/stage is assumed (Section 4, Assumption 4.8). This gives a combined efficiency of 99.99 percent for the two stages; a HEPA LPF of 10^{-4} (Section 4, Assumption 4.8). It is further assumed that the HVAC system is removing particulates and cesium in air through two stages of HEPA filters in series, which are protected by prefilters, sprinklers, and demisters. NUREG/CR-0722 test data (Lorenz et al. 1980, Table 19) show that two-stage HEPA filters capture almost 100 percent of incoming airborne cesium. For normal operations and Category 1 event sequences, no credit is taken for charcoal adsorbers to remove gaseous radionuclides.

3.1.3.2 Worker Dose From Resuspension of Surface Contamination, Activated Air, or Activated Silica Dust

Equations 3, 4, 8, and 9, are applicable to the worker dose from: (1) resuspension of surface contamination from transportation casks or waste packages inside a waste transfer cell, or (2) exposure to neutron-activated air or silica dust from emplacement drifts.

Equation 7 is not used and the term $ST_j \times FA$ in Equations 5 and 6 is replaced with ST_j, where ST_j is the total release source term from cask or waste package surface contamination and activated air and activated silica dust from the emplacement drifts. The ST_j is calculated in BSC (2004g, Table 14) and is summarized in Section 6.1.2.2.

3.1.3.3 Shielding Source Term Description

3.1.3.3.1 Shielding Source Term Criteria

The following shielding source term criteria are used for the shielding design for the surface and subsurface facilities (BSC 2004n, Section 4.9.1.4):

- Shielding source terms for the surface and subsurface facility design are based on the limiting waste form, as well as the limiting waste package type.
- Design basis and maximum source terms are established to provide sufficient and bounding coverage of the historical and projected fuel inventory for normal operations and Category 1 event sequences. The design basis source term covers a minimum of 95 percent of the total inventory, with provisions made to accommodate the remaining 5 percent. The maximum source term represents the bounding SFA in the inventory at the repository. The design basis or average source terms are used for normal operations and Category 1 event sequences. The maximum source terms are used for Category 2 event sequences.
- Minimum initial enrichment is established as stated in ISG-6 (NRC 2001) for the selected SFA used to determine the source term, because lower enriched fuel irradiated to the same burnup as higher enriched fuel produces a higher source.

3.1.3.3.2 Limiting Waste Form

The repository receives, handles, processes, packages, and emplaces a variety of waste forms including commercial SNF from PWR and BWR power plants, high-level radioactive waste (HLW), U.S. Department of Energy (DOE) SNF, and naval SNF. A source comparison study determined that a single PWR SNF assembly with a 75-GWd/MTU burnup, a 5-wt percent initial enrichment, and a 5-year cooling time is a limiting radiation source for canistered SNF and HLW glass canisters as described in *Shielding Calculation for Dry Transfer Facility, Remediation Facility, and Canister Handling Facility* (BSC 2004o, p. 19). Therefore, PWR SNF represents the limiting waste form and is selected for specification of shielding source terms for repository design.

3.1.3.3.3 Limiting Waste Package

Several dose rate calculations have been performed for different capacity waste packages, including:

- 21 PWR uncanistered fuel waste package
- 44 BWR uncanistered fuel waste package
- HLW or DOE SNF codisposal waste package, or both
- Naval SNF waste package.

Other waste package types, such as the 12 PWR and 24 BWR small waste packages, are less radiation limiting than the corresponding large waste packages as described in *Subsurface Shielding Source Term Specification Calculation* (BSC 2002, p. 21).

Comparison of the dose rate calculations for the different waste packages shows that the 21 PWR waste package has the highest dose rate on the surface of the waste package and, thus, the 21 PWR is the limiting waste package for shielding analysis (BSC 2002, p. 22).

3.1.3.3.4 Maximum Fuel Specification

With PWR SNF as the limiting waste form, the limiting SFA is a PWR SFA with the following characteristics, as described in *PWR Source Term Generation and Evaluation* (BSC 2004l, p. 27):

- 5 percent initial ^{235}U enrichment
- 80 GWd/MTU burnup
- 5 years cooling time.

Shielding calculations use the maximum source terms for operations involving a single assembly; examples include fuel transfer from the transportation cask through the transfer cell to the waste package, fuel transfer to the staging area, and fuel transfer in the remediation pool.

3.1.3.3.5 Design Basis Fuel Specification

The design basis source terms shall cover a minimum of 95 percent of the total historical and projected fuel inventory (BSC 2004n, Section 4.9.1.4). The design basis fuel specification provides the shielding source terms for a loaded waste package.

The waste package source specification uses the following parameters as a design basis (BSC 2002, Section 5.5.2), because the waste package thermal output is limited to 11.8 kW:

- 4 percent initial ^{235}U enrichment
- 60 GWd/MTU
- 10 years cooling.

These characteristics result in a heat generation rate of 1.2 kW/SFA, which is more than twice the average heat output per assembly allowed in the 21 PWR waste package.

3.1.3.3.6 Axial Power Peaking Factor

Source terms in Tables 6 and 7 represent the values for uniform fuel burnup; a relative power factor of 1 for the entire length of the SFA. Shielding source terms account for the power peaking factor to obtain the peak dose rate, because fuel burnup varies axially along the length of the SFA.

Table 6. Maximum and Design Basis Pressurized Water Reactor Spent Nuclear Fuel Neutron Source Terms

Neutron Energy Range (MeV)		Neutron Source (n/s per assembly)	
Upper Bound	Lower Bound	Maximum[a]	Design Basis[b]
2.00E+01	6.43E+00	3.93E+07	1.54E+07
6.43E+00	3.00E+00	4.43E+08	1.74E+08
3.00E+00	1.85E+00	4.85E+08	1.91E+08
1.85E+00	1.40E+00	2.76E+08	1.09E+08
1.40E+00	9.00E-01	3.76E+08	1.48E+08
9.00E-01	4.00E-01	4.11E+08	1.61E+08
4.00E-01	1.00E-01	8.05E+07	3.16E+07
1.00E-01	1.70E-02	0.00E+00	0.00E+00
1.70E-02	3.00E-03	0.00E+00	0.00E+00
3.00E-03	5.50E-04	0.00E+00	0.00E+00
5.50E-04	1.00E-04	0.00E+00	0.00E+00
1.00E-04	3.00E-05	0.00E+00	0.00E+00
3.00E-05	1.00E-05	0.00E+00	0.00E+00
1.00E-05	3.05E-06	0.00E+00	0.00E+00
3.05E-06	1.77E-06	0.00E+00	0.00E+00
1.77E-06	1.30E-06	0.00E+00	0.00E+00
1.30E-06	1.13E-06	0.00E+00	0.00E+00
1.13E-06	1.00E-06	0.00E+00	0.00E+00
1.00E-06	8.00E-07	0.00E+00	0.00E+00
8.00E-07	4.00E-07	0.00E+00	0.00E+00
4.00E-07	3.25E-07	0.00E+00	0.00E+00
3.25E-07	2.25E-07	0.00E+00	0.00E+00
2.25E-07	1.00E-07	0.00E+00	0.00E+00
1.00E-07	5.00E-08	0.00E+00	0.00E+00
5.00E-08	3.00E-08	0.00E+00	0.00E+00
3.00E-08	1.00E-08	0.00E+00	0.00E+00
1.00E-08	1.00E-11	0.00E+00	0.00E+00

NOTES: [a] BSC 2004l, Attachment X; PWR fuel with 5 percent enrichment, 80 GWd/MTU burnup and 5-year cooling
[b] BSC 2004l, Attachment X; PWR fuel with 4 percent enrichment, 60 GWd/MTU burnup and 10-year cooling.

MeV = million electron volt; n/s = neutrons per second.

Table 7. Maximum and Design Basis Pressurized Water Reactor Spent Nuclear Fuel Gamma Source Terms

Gamma Energy Range (MeV) Upper – Lower		Maximum Source (γ/s per assembly)[a]				Design Basis Source (γ/s per assembly)[b]			
		Fuel Region	Bottom Region	Plenum Region	Top Region	Fuel Region	Bottom Region	Plenum Region	Top Region
5.00E-02	1.00E-02	2.33E+15	5.94E+11	5.28E+11	3.79E+11	1.21E+15	2.73E+11	1.88E+11	1.75E+11
1.00E-01	5.00E-02	6.44E+14	1.16E+11	6.09E+10	7.43E+10	3.29E+14	5.28E+10	2.77E+10	3.39E+10
2.00E-01	1.00E-01	5.22E+14	2.83E+10	3.52E+10	1.79E+10	2.45E+14	1.28E+10	1.17E+10	8.19E+09
3.00E-01	2.00E-01	1.48E+14	1.41E+09	1.96E+09	8.91E+08	7.13E+13	6.39E+08	6.33E+08	4.07E+08
4.00E-01	3.00E-01	9.85E+13	1.90E+09	5.86E+09	1.17E+09	4.55E+13	8.50E+08	1.64E+09	5.33E+08
6.00E-01	4.00E-01	1.53E+15	1.91E+09	1.10E+11	7.41E+07	2.26E+14	4.92E+08	2.69E+10	3.37E+07
8.00E-01	6.00E-01	4.70E+15	4.35E+09	5.95E+10	2.37E+09	2.37E+15	2.91E+09	1.60E+10	1.86E+09
1.00E+00	8.00E-01	7.08E+14	1.37E+11	8.03E+09	7.66E+10	1.22E+14	5.40E+09	2.48E+09	3.41E+09
1.33E+00	1.00E+00	4.55E+14	3.38E+13	1.74E+13	2.17E+13	1.95E+14	1.54E+13	7.97E+12	9.90E+12
1.66E+00	1.33E+00	1.30E+14	9.53E+12	4.91E+12	6.12E+12	4.50E+13	4.35E+12	2.25E+12	2.80E+12
2.00E+00	1.66E+00	1.44E+12	1.87E+03	9.19E+02	1.13E+03	1.52E+11	2.35E+00	1.49E+02	2.15E-02
2.50E+00	2.00E+00	2.49E+12	2.26E+08	1.16E+08	1.45E+08	5.17E+10	1.03E+08	5.34E+07	6.64E+07
3.00E+00	2.50E+00	1.10E+11	3.51E+05	1.81E+05	2.25E+05	3.79E+09	1.60E+05	8.29E+04	1.03E+05
4.00E+00	3.00E+00	1.39E+10	7.66E-08	1.00E-08	4.16E-08	4.97E+08	9.43E-10	1.55E-10	5.19E-10
5.00E+00	4.00E+00	7.09E+07	0.00E+00	0.00E+00	0.00E+00	2.82E+07	0.00E+00	0.00E+00	0.00E+00
6.50E+00	5.00E+00	2.85E+07	0.00E+00	0.00E+00	0.00E+00	1.13E+07	0.00E+00	0.00E+00	0.00E+00
8.00E+00	6.50E+00	5.58E+06	0.00E+00	0.00E+00	0.00E+00	2.22E+06	0.00E+00	0.00E+00	0.00E+00
1.00E+01	8.00E+00	1.19E+06	0.00E+00	0.00E+00	0.00E+00	4.71E+05	0.00E+00	0.00E+00	0.00E+00

NOTES: [a] BSC 2004l, Attachment X; PWR fuel with 5 percent enrichment, 80 GWd/MTU burnup and 5-year cooling
[b] BSC 2004l, Table 10; PWR fuel with 4 percent enrichment, 60 GWd/MTU burnup and 10-year cooling.

MeV = million electron volt; γ/s = photons per second.

For PWR fuel, shielding calculations use a conservative peaking factor of 1.25, based on the predicted heat profile for a PWR assembly provided in *The TN-24P PWR Spent-Fuel Storage Cask: Testing and Analyses* (Creer et al. 1987, p. 3-29). This factor is directly applied to the calculated gamma dose rate as a multiplier; the gamma source strength is reasonably proportional to fuel burnup. The multiplier for the neutron dose rate is determined to be the ratio of the neutron source strength at the peak burnup to that for uniform burnup, as described in *Dose Rate Calculation for 21-PWR Waste Package* (BSC 2004p, Section 5.2.1).

3.1.3.4 Shielding Methodologies

3.1.3.4.1 Shielding Computer Codes

The shielding analysis to support the repository facility design uses methods and codes that are consistent with those of commonly accepted shielding calculations and that are appropriate for radiation types, geometry, and materials. The analytical tools include codes that use Monte Carlo, deterministic transport, and point-kernel integration techniques for the various shielding problems encountered in the repository design.

Simple and scoping-type gamma shielding problems are handled with the efficient point-kernel integration codes.

Complex or deep-penetration shielding problems require the use of Monte Carlo or deterministic transport codes, especially for problems involving neutron and secondary gamma dose contributions.

The following shielding codes have been benchmarked, validated, qualified, and baselined in accordance with LP-SI.11Q, *Software Management* to support the repository design:

- MCNP from *MCNP-A General Monte Carlo N-Particle Transport Code* (Briesmeister 1997)
- SCALE V4.3 from *SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation* of NUREG/CR-0200 (ORNL 1997)
- QAD-CGGP from *Final Version Description Document for the QAD-CGGP Computer Code* (CRWMS M&O 1995); a three-dimensional point-kernel gamma transport shielding computer code
- PATH from *PATH Gamma Shielding Code User's Manual* (Su et al. 1987); a three-dimensional point-kernel gamma shielding computer code.

Except for PATH (Su et al. 1987), the NRC recognizes the acceptance of these codes for analysis of SNF storage facilities in NUREG-1567 (NRC 2000, pp. 7-12 and 7-13). PATH is similar to the accepted QAD-CGGP code and is accepted by the NRC in the GA-4 cask certification, *GA-4 Legal Weight Truck Spent Fuel Shipping Cask, Safety Analysis Report for Packaging (SARP)* (General Atomics 1998, p. 5.4-2).

MCNP (Briesmeister 1997) is a general purpose Monte Carlo code for neutron, photon, or coupled neutron-photon transport problems, suitable for complex three-dimensional geometry and a variety of radiation source types. MCNP represents the best choice for Monte Carlo shielding analysis, because it is widely used in the nuclear industry for various applications.

SCALE V4.3, from NUREG/CR-0200 (ORNL 1997), contains the modules for performing source term and shielding calculations. For Yucca Mountain applications, SCALE can be used for both source term and shielding calculations. The ability of updating burnup-dependent cross section library in SCALE provides a more accurate determination of the source terms than the ORIGEN code provided in *RSIC Computer Code Collection, ORIGEN 2.1, Isotope Generation and Depletion Code, Matrix Exponential Method* (ORNL 1991).

QAD-CGGP (CRWMS M&O 1995) and PATH (Su et al. 1987) are both point-kernel integration codes for gamma shielding problems, capable of explicitly treating three-dimensional source shield configurations. These two codes are similar in capabilities and produce results in satisfactory agreement. PATH provides additional features to treat multiple sources and various source types in a single run.

Flux-to-DCFs (BSC 2002, Section 5.6.1) used in the shielding analyses are consistent with ANSI/ANS-6.1.1-1977, pp. 4 and 5. No degradation or loss of shielding materials from an event sequence has been identified.

The computer codes listed in this section are not used to produce results in this calculation.

3.1.3.5 Worker Dose From Direct Radiation

During the preclosure period, some surface and subsurface facility workers could be exposed to direct radiation when work requires them to be in close proximity to contained sources, such as transportation casks, storage casks, or waste packages. If necessary, fixed or portable shielding materials are used to reduce the dose rates to which workers are exposed. The dose rate at a distance from the contained source was calculated using the MCNP computer program (Briesmeister 1997). The dose assessment involves calculations of annual individual doses to workers. The dose contributions from contained radiation sources, such as SFAs in a transportation cask or waste package, are obtained from the shielding calculations using NRC accepted computer codes, such as MCNP. These shielding calculations generate dose maps around functional areas within the facilities for determining dose rates at worker locations.

Dose assessments are performed by job function or worker group using time-motion inputs and dose rates calculated at worker locations. The time-and-motion inputs define the process step, location, number of workers, and duration of worker occupancy. The individual doses are calculated for each process step and summed over the process steps to obtain cumulative external exposures to workers on an annual basis. Outputs of the assessment consist of a matrix of operations, locations, source, frequency, dose rates, stay times, and doses. In addition, to demonstrate regulatory compliance, calculated annual individual doses are used for comparison with the ALARA design goal to minimize the number of individuals who have the potential of receiving more than 500 mrem/year TEDE (BSC 2004n, Section 4.9.3.3).

The Transportation Cask Receipt and Return Facility (TCRRF) provides space, layout, and structures that support waste handling operations. Waste handling operations, including unloading transportation casks from the transportation carrier, integrate with the onsite cask-handling system within the TCRRF protective structure to support the throughput rates established for waste emplacement.

TCRRF worker groups include: transportation personnel, cask operator, health physicist, and gantry/crane operator. Transportation personnel duties include the movement of a full or empty cask/carrier into and out of the transfer bay. The cask operator and gantry/crane operator work together and perform duties that involve removal of personnel barriers/impact limiters, cask transfers from a carrier to a site rail transfer cart, and inspection of casks. The health physicist monitors radiation and contamination levels during cask receipt and transfer.

The DTF provides space, layout, structures, and equipment that transfer uncanistered SFAs from transportation casks to waste packages and prepare sealed waste packages for subsurface transport and emplacement. Waste handling activities include transportation cask and waste package receipt and processing and transfer bay operation. DTF worker groups include: the cask and waste receipt operator, transfer bay operator, and health physicist.

The FHF is a surface facility supporting waste handling operations. The facility receives transportation casks, unloads and transfers their contents to waste packages or site specific casks, transfers the site specific casks to the aging area, prepares waste packages, and transfers waste packages to a transporter for subsurface emplacement. FHF worker groups include: the cask and waste receipt operator, transfer bay operator, and health physicist.

The annual external dose from the contained source in casks per individual worker is calculated as (BSC 2004j, Equation 1):

$$ED_g = \sum_i [DR_{i,g} \times T_i] \times (TC / WS) \qquad \text{(Eq. 10)}$$

where,

ED_g	=	Annual external dose per individual worker per work group g (mrem/year)
$DR_{i,g}$	=	External dose rate at location i per individual worker per work group g (mrem/hr)
T_i	=	Duration of the exposure at location i, per cask operation (hr/cask)
TC	=	Number of processed transportation casks per year (casks/year)
WS	=	Number of work shifts.

Worker locations for cask preparation operations are categorized in terms of distances from the transportation cask surface. The distances are estimated from the most likely worker locations to perform the specific tasks for the cask preparation operations. For hands-on activities, such as swipes for surface contamination sampling, the worker is assumed to be 1 m from the cask. For processing tasks that are not hands-on, but require worker presence in the area, the worker is assumed to be standing at a reasonable distance from the cask. For processing transportation casks, workers are assumed to be at one of three distances from the exterior surfaces of a transportation cask: 1 m, 5 m, or 10 m. Dose rates in the vicinity of a transportation cask are estimated using the 2-m dose rates of the Universal Transport Cask and the radial dose distance factors derived from the TN-32 cask (BSC 2004c, p. 11). Average dose rates from the cask surface are: 9.10 mrem/hr at 1 m, 1.66 mrem/hr at 5 m, and 0.51 mrem/hr at 10 m (BSC 2004c, p. 11).

The equation used to calculate the annual dose to a waste package remediation worker or to a waste package emplacement and retrieval worker for DTF 1, DTF 2, the FHF, the Remediation Facility, and the subsurface repository (BSC 2004i, Equation 1), is:

$$ED_g = \sum_i [DR_{i,g} \times T_i] \times WP_g \qquad \text{(Eq. 11)}$$

where,

ED_g	=	Annual external dose per individual worker per work group g (mrem/year)
$DR_{i,g}$	=	External dose rate at location i, per individual worker per work group g (mrem/hr)
T_i	=	Duration of the exposure at location i, per transfer operation (hr/transfer)
WP_g	=	Annual number of waste package transfers involving a worker in group g per year (transfers/year).

To assess the individual dose to maintenance workers, Equation 11 is modified as follows (BSC 2004i , Equation 2):

$$ED_g = \sum_i [DR_{i,g} \times T_i] \times M_g \qquad \text{(Eq. 12)}$$

where,

ED_g	=	Annual external dose per individual worker per work group g (mrem/year)
$DR_{i,g}$	=	External dose rate at location i, per individual worker per work group g (mrem/hr)
T_i	=	Duration of the exposure at location i, per maintenance operation (hr/operation)
M_g	=	Annual number of maintenance operations involving a worker in group g per year (operations/year).

Dose Aggregation–Total Category 1 annual worker dose is based on contributions from four sources:

- Category 1 event sequences
- Normal operational releases from surface facilities
- Normal operational releases from the subsurface repository
- Direct radiation dose from contained radiation sources.

Equation 1 is slightly modified to yield the total Category 1 annual worker dose, D_{TOT}:

$$D_{TOT} = D_{Cat.1} + D_{NO}^{SF} + D_{NO}^{Sub} + D_{No}^{Dir} \qquad \text{(Eq. 13)}$$

where,

$D_{Cat.1}$	=	Annual dose from Category 1 event releases (mrem/year)
D_{NO}^{SF}	=	Annual dose from normal operational releases from surface facilities (mrem/year)
D_{NO}^{Sub}	=	Annual dose from normal operational releases from the subsurface repository (mrem/year)
D_{NO}^{Dir}	=	Annual dose from direct radiation from contained sources (mrem/year).

Worker dose assessment includes individual doses for both normal operations and Category 1 event sequences. Total dose, including internal and external exposures, is calculated on an annual basis by summing the contributions from normal operations and frequency-weighted doses from Category 1 event sequences for demonstration of regulatory compliance. Equation 2 is used to calculate annualized worker dose from Category 1 event sequences.

Worker dose assessment includes contributions from both surface and subsurface facilities on a building-by-building basis; the assessment of subsurface facilities also includes contributions from operations and activities performed underground.

3.2 METHODOLOGY FOR CATEGORY 2 EVENT SEQUENCES

This section discusses the methodology used for evaluating radiological consequences from Category 2 event sequences.

3.2.1 Category 2 Event Sequence Performance Objectives

The four dose measures of LDE, SDE, TEDE, and TODE used to evaluate normal operations and Category 1 event sequences are also used to evaluate Category 2 event sequences. Performance objectives (Section 6.2.3) for Category 2 event sequences are summarized in Table 2.

3.2.2 Category 2 Event Sequence Public Dose Methodology

In Category 2 event sequences, radioactive materials are assumed to be released as a ground-level radioactive plume (Section 4, Assumption 4.17) and the building wake effect is considered. The plume is dispersed en route to the site boundary, resulting in an acute individual exposure during plume passage and a chronic individual exposure to ground contamination and contaminated food for 1 year after plume passage (Section 4, Assumption 4.5). No credit is taken for plume meandering or for plume rise caused by initial thermal energy to further ensure that the χ/Q value is conservative at the site boundary. Potential radiation doses from inhalation, resuspension inhalation, ingestion, air submersion, and groundshine pathways are considered for Category 2 event sequences. Inhalation, resuspension inhalation, ingestion, air submersion, and groundshine doses are calculated using MACCS2 (ORNL 1998).

The χ/Q values are calculated internally by MACCS2 (ORNL 1998) using hourly meteorological data gathered at Yucca Mountain from 1998 through 2002. The 1999 meteorological data are used because the data generate conservative χ/Q values (Section 6.2.1.1.2). The χ/Q value output from MACCS2 (ORNL 1998) for surface releases is a mean, 50th percentile, 95th percentile, or 99.5th percentile evaluated at 11 km for surface releases (Section 6.2.1.5 and Attachment B, Figure B-1).

Category 2 event sequences use the 95th percentile with a 5 percent exceedance of the χ/Q value evaluated in accordance with Regulatory Guide 1.145, Section 3, at the closest site boundary of 11 km for the maximally exposed individual without regard to sectors.

Commercial SNF Source Term–Source terms for PWR and BWR SFAs with four different combinations of initial enrichment, burnup, and decay time are considered. These combinations are presented in Table 3. Radionuclide inventories in Ci/FA for Maximum PWR and Maximum BWR SFAs are presented in Table 4. For Category 2 event sequences, both Maximum PWR and Maximum BWR SFAs are used to calculate maximum doses; the 95th percentile dose in accordance with Regulatory Guide 1.145, Section 3.

The ARFs and RFs for intact commercial SNF rods (Table 5) are used for Category 2 event sequences, because the combined ARF × RF values for intact commercial SNF rods are larger or equal to the values for intact commercial SNF assemblies.

Naval SNF Source Term–A bounding radionuclide release source term was developed which covers all naval SNF canister inventories and internal hardware designs. Conservative estimates of SNF inventory, crud inventory, fuel damage ratio, and release fractions were used to develop a bounding radionuclide release source term which will result in doses larger than those expected should an actual Category 2 event sequence occurs.

For naval SNF assemblies, a conservative radionuclide inventory for a canister of naval SNF at five years after shutdown is developed using Naval Nuclear Propulsion Program depletion codes in conjunction with ORIGEN-S, which have been qualified for use with naval with naval SNF in repository applications. The type of SNF assembly that is used in the analysis is the one that represents a majority of the naval SNF assemblies that will be emplaced. The inventory assumed up to 10 percent more SFAs in a canister that will be loaded.

The crud contribution to the source term for naval SNF is calculated using the standard naval program shielding procedure. This value is increased by a factor of 2.5 to provide a conservative crud concentration at core shutdown for use in developing the source term.

The fuel and crud damage ratios used to developed the source term were conservatively selected to ensure that all canister radionuclide inventories and internal hardware designs are bounded. The ARFs and RFs for intact commercial SNF assemblies (Table 5) are used for Category 2 event sequences involving naval SNF.

HLW Source Term–Bounding per-canister HLW radionuclide data were obtained from four different DOE sites, including the Savannah River Site, Hanford Site, West Valley, and Idaho National Engineering and Environmental Laboratory (INEEL). Projected bounding per-canister radionuclide inventory data for nuclides were set equal to the maximum values from sludge batches of wastes, vitrified between October 10, 1994 and March 17, 2003, as presented in *Projected Glass Composition and Curie Content of Canisters from the Savannah River Site (U)* (Fowler 2003, Section 5.0). The methodology used for Category 2 event sequences involving HLW is to demonstrate that the Savannah River Site HLW bounds HLW from the other DOE sites.

In summary, the methodology for Category 2 event sequence public dose calculations is to maximize the public dose to account for the variability or uncertainty, or both, in MAR, source terms, release fractions, and varying weather conditions. Weather conditions relevant to dose calculations include wind speed, wind direction, stability class, and precipitation.

MACCS2 (ORNL 1998) randomly samples the Yucca Mountain site-specific hourly data over a 1-year period, producing 8,760 sets of wind speed, wind direction, stability class, and precipitation to calculate χ/Q values.

Each set of weather data calculates one value of χ/Q. Depending of the number of sampling chosen, thousands of χ/Q values could be generated in one MACCS2 (ORNL 1998) run. Statistical analysis is performed internally in MACCS2 on these χ/Q values. Values for the mean, 50th percentile, 95th percentile, and 99.5th percentile are generated and printed on the MACCS2 (ORNL 1998) output file.

4. ASSUMPTIONS

The following assumptions, except for Assumption 4.14, are used for calculating public dose; Assumption 4.14 is used in calculations in other documents for calculating worker dose.

4.1 Waste forms involved in normal operations and in Category 1 and Category 2 event sequences include PWR or BWR SFAs, HLW, and naval SNF.

Basis: The only waste form not included is DOE SNF, because a drop and breach of a DOE SNF canister, including the 18 in. or 24 in. standardized canister or the multicanister overpack is a beyond Category 2 event sequence (BSC 2004k, Section 4.2.3).

This assumption is used in Section 6.1.

4.2 At the highest nominal receipt rate, 3000 MTHM (DOE 2004b, Table 1), of commercial SNF pass through the CHF, DTF 1, DTF 2, and the FHF each year. It is assumed that for the purposes of calculating worker and public doses, fuels received are PWR SFAs. Using an average PWR assembly weight of 0.475 MTHM/SFA (BSC 2004l, p. 8), the highest nominal throughput is 6,316 SFAs/year.

Basis: It is conservative to assume that 3,000 MTHM of commercial SNF is PWR SFAs, because PWR SFAs typically have higher enrichment and burnup than BWR SFAs (Table 3). SNF with a higher enrichment and a higher burnup, in general, has a higher total radionuclide inventory.

This assumption is used in Section 6.1.2.1.

4.3 One percent (i.e., DR of 0.01) of the fuel rods received at the repository are modeled as having defect sizes equal to pinhole leaks or hairline cracks and the fission product gases, volatile species, and fuel fines are released. Releases from one percent of the 6,316 SFAs are used as the source term for the calculation of normal operations doses.

Basis: The 1 percent fuel breakage assumption is from ISG-5 (NRC 2003a, Attachment, p. 7) for normal operations. Historical fuel discharge data have shown that the fuel rod failure rate is about 0.01 percent, versus the range of 0.02 to 0.07 percent in the first 20 years of commercial nuclear power, as described in *2002 Waste Stream Projections Report* (BSC 2003c, p. G-6). These numbers are much smaller than the 1 percent fuel failure rate.

This assumption is used in Sections 3.1.3.1 and 6.1.2.1.

4.4 The HEPA filters of the surface facility HVAC systems are assumed to be unavailable to remove radionuclides for Category 2 event sequences. HEPA filters are assumed to be functioning for surface facility normal operations and Category 1 event sequence dose calculations.

Basis: It is conservative to not take credit for the HEPA filters to mitigate Category 2 event sequences. Credit is taken for the HEPA filters to mitigate normal operational releases and Category 1 event sequences.

This assumption is used in Section 6.1.4, 6.1.5, and 6.2.1.3.

4.5 It is assumed that for Category 1 and Category 2 event sequences, radioactive materials are released in a 1-hr duration.

Basis: The 1-hr release duration is conservative when compared with the 2-hr χ/Q values specified in Regulatory Guide 1.145, Section 1.3.

This assumption is used in Sections 3.1.2, 3.2.2, 6.1.4, 6.1.5, and 6.2.1.5.

4.6 For calculating public doses from normal operations using MACCS2 (ORNL 1998), the release duration is assumed to be 24 hr, which is the longest duration allowed by MACCS2. The release is assumed to result in an acute individual exposure during plume passage and a chronic individual exposure to ground contamination and contaminated food after plume passage. The period of long-term exposure to ground contamination and intake of contaminated food is 1 year.

Basis: The sensitivity run discussed in Section 6.3.4.1 showed that the use of the release duration of 24 hr, instead of 1 year, for normal operations is conservative.

This assumption is used in Sections 3.1.2, 6.2.1.2, 6.2.1.5, and 6.3.4.1.

4.7 It is assumed that radionuclides are released from surface facilities during normal operations via the exhaust stack.

Basis: The CHF, DTF 1, DTF 2, and the FHF have a vent that is higher than the adjacent structures. For surface facility normal operations, credit is taken for the elevated release to disperse radionuclides in the air.

This assumption is used in Sections 3.1.2 and 6.3.

4.8 For normal operations and Category 1 event sequences, a two-stage HEPA filtration system with a particulate removal efficiency of 99 percent per stage (i.e., a HEPA leak path factor $(LPF)_{HEPA}$ of 0.01/stage) is assumed. This gives a combined efficiency of 99.99 percent for two stages; a HEPA LPF of 10^{-4}. It is further assumed that the HVAC system is removing particulates and cesium in air through two stages of HEPA filters in series that are protected by pre-filters, sprinklers, and demisters.

Basis: A HEPA filter LPF of 10^{-4} is more conservative than the value of 2×10^{-6} for two HEPA filters, as presented in NUREG/CR-6410 (SAIC 1998, Section F.2.1.3). NUREG/CR-0722 test data (Lorenz et al. 1980, Table 19) shows that two-stage HEPA filters capture almost 100 percent of incoming airborne cesium. NUREG/CR-0722 test data is applicable because the test was conducted at a temperature of 900°C (Lorenz et al. 1980, p. 48), which conservatively bounds SNF temperatures inside a waste transfer cell under normal or accident conditions.

This assumption is used in Sections 3.1.3.1 and 6.2.1.3.

4.9 Drops or collisions of canistered single commercial SNF rods in transportation casks and waste packages are assumed to be Category 2 event sequences.

Basis: Drops or collisions of commercial SNF in transportation casks and waste packages are Category 2 event sequences (BSC 2004k, Section 7). Canistered single commercial SNF rods are only a subset of commercial SNF. Therefore, the frequency of drops or collisions of canistered single commercial SNF fuel rods in transportation casks and waste packages is lower than the frequency of drops or collisions of commercial SNF in transportation casks and waste packages. Thus, drops or collisions of canistered single commercial SNF rods in transportation casks and waste packages are Category 2 event sequences.

This assumption is used in Sections 3.1.2 and 6.2.1.3.

4.10 For Category 1 and Category 2 event sequences, the DR is assumed to be 1.0 for commercial SNF. For Category 2 event sequences, the DR is assumed to be 1.0 for HLW in a canister. The DR is the fraction of fuel rods that is assumed to fail by cladding breach during an event sequence or the fraction of HLW that is damaged by crush or impact, or both. Bounding damage ratios are used for Category 2 event sequences involving naval SNF, as documented in NNPP Input for Yucca Mountain Project Preclosure Safety Analyses (Gisch 2004, p. 1).

Basis: These are bounding assumptions for commercial SNF, HLW, and naval SNF.

This assumption is used in Sections 3.1.3.1, 6.1.4, and 6.2.1.3.

4.11 For Category 2 event sequences, the cask leak path factor $(LPF)_{cask}$ is assumed to be 0.1 for SNF in a transportation cask and 0.01 for HLW in a canister in a transportation cask, and 1.0 for naval SNF in a transportation cask or in a canister. The $(LPF)_{cask}$ is the fraction of the ARF that reaches the ventilation system after local deposition, consisting of plate-out and gravitational settling, within a transportation cask.

Basis: This is a conservative assumption. For rail casks involved in end, corner, or side impacts, at an impact speed of 60 mph, NUREG/CR-6672 (Sprung et al. 2000, Table 7.19) calculated an $(LPF)_{cask}$ of 0.02 for particles and 0.0008 for cesium, which are smaller than the conservatively assumed $(LPF)_{cask}$ of 0.1. Both the canister and the cask are assumed to have an LPF of 0.1 because HLW is shipped inside of a canister within a transportation cask. The combined LPF for HLW is equal to 0.01 (i.e., $0.1 \times 0.1 = 0.01$). An $(LPF)_{cask}$ of 1.0 for naval SNF is conservative.

This assumption is used in Section 6.2.1.3.

4.12 For normal operations and Category 1 event sequences, the facility leak path factor $(LPF)_{fac}$ is conservatively assumed to be 1.0 inside a waste transfer cell. The $(LPF)_{fac}$ is the fraction of the ARF that reaches the ventilation system after local deposition, consisting of plate-out and gravitational settling, within a surface facility.

Basis: This is a bounding assumption because no credit is taken for local deposition of particulates within a surface facility.

This assumption is used in Section 6.2.1.3.

4.13 It is assumed that the HVAC system is operating and no airborne material released from Category 1 event sequences leaks into space occupied by workers who work in rooms adjacent to a waste transfer cell in a DTF or the FHF.

Basis: This assumption is reasonable because the transfer cell confinement and the HVAC system are designed to prevent any leakage to rooms adjacent to the transfer cell in the event of a Category 1 event sequence.

This assumption is used in Sections 6.1.2.1 and 6.1.4.

4.14 For normal operations and Category 1 event sequences, it is assumed that for radionuclides released from a waste transfer cell within a surface facility, the HVAC system is operating and airborne radionuclides are vented through the building exhaust stack, dispersed into the atmosphere, and then reenter the building through the building ventilation system air intakes. It is assumed that for radionuclides released from the subsurface facility, airborne radionuclides are dispersed into the atmosphere and reenter the subsurface facility through the subsurface ventilation system air intakes.

Basis: The calculation of worker doses considers that the concentrations within the facility are at equilibrium with the concentration outside of the air intake, which is the major source of air into the facility. Other pathways, such as through the entrance air locks would contribute concentrations at lower levels than at the air intake because the dispersion at these locations will be greater than at the air intake. Therefore, these pathways were conservatively neglected. The subsurface ventilation system air intakes are the only places that airborne radionuclides could enter the subsurface facility.

This assumption is used in Section 3.1.3.1.

4.15 The maximally exposed individual is defined as an individual located at a distance that corresponds to the approximate distance between the surface facility or the subsurface repository and the nearest point of public access on the repository site boundary, which lies to the west. The proposed Land Withdrawal Area boundary (Attachment B, Figure B-1) is assumed to be the site boundary (i.e., preclosure controlled area boundary). A site boundary distance of 11 km (Attachment B, Figure B-1) is used to calculate χ/Q values from radiological releases from the surface facility. This distance corresponds to the distance from the DTF ventilation exhaust shaft to the nearest point on the site boundary that is the closest point where any member of the public could be standing or living at the time of a postulated radiological release. A site boundary distance of 8 km (Attachment B, Figure B-1) is used to calculate χ/Q values from radiological releases from the subsurface repository. This distance corresponds to the approximate distance between the subsurface repository and the nearest point of public access on the site boundary, which lies to the west.

Basis: The use of the shortest distances of approximately 11 km and 8 km from the surface and subsurface release points, respectively, to the site boundary to calculate the public dose is conservative.

This assumption is used in Sections 6.2.1.5, 6.3.3, and 6.3.5.

4.16 It is assumed that the fission product gas, volatile species, and crud release fractions for breaches of intact commercial SNF assemblies and rods in Table 5 are applicable to releases of fission product gases, volatile species, and crud during oxidation of damaged commercial SNF in air in Table 5 (BSC 2004m, Section 6.2.1.3 and Table 11).

Basis: The volatile species release fractions for breaches of intact commercial SNF assemblies and rods given in Table 5 bound the data on ^{137}Cs and ^{106}Ru release fractions during oxidation of damaged commercial SNF in air (i.e., 1.4×10^{-5} for ^{137}Cs, and 7.27×10^{-6} for ^{106}Ru) from NUREG/CR-6672 (Sprung et al. 2000 p. 7-45). The 30 percent release fraction for gases for breaches of intact commercial SNF assemblies and rods given in Table 5 bounds the fission product gas release fractions during oxidation of damaged commercial SNF in air given in "Effects of an Oxidizing Atmosphere in a Spent Fuel Packaging Facility" (Einziger 1991, Figure 8). Because the crud is deposited on the outer surfaces of commercial SNF assemblies or rods, the release fraction is expected to be the same for both intact and damaged commercial SNF.

This assumption is used in Section 6.2.1.3.

4.17 The releases from the subsurface exhaust shafts during normal operations and the releases from the surface facilities from Category 1 and Category 2 event sequences are assumed to be at ground level.

Basis: An occurrence of a ground-level release of radioactive material, instead of an elevated release from the subsurface exhaust shafts, is conservative. Releases from Category 1 event sequences are assumed to exit surface facilities through a stack and are conservatively modeled as ground level releases. Releases from Category 2 event sequences are also modeled as ground level releases.

This assumption is used in Sections 3.1.2, 3.2.2, 6.1.2.1, 6.1.2.2, 6.1.4, and 6.3.

4.18 For normal operations, and Category 1 and Category 2 event sequences, no credit is taken for charcoal adsorbers to remove radionuclides.

Basis: Currently there is no plan to install charcoal adsorbers. If in the future charcoal adsorbers are installed, this assumption would result in a slightly higher dose and, therefore, it is a conservative assumption.

This assumption is used in Section 6.2.1.3.

4.19 It is assumed that the canister handling system is designed so that a drop of an HLW canister inside a surface facility will not exceed a drop height of 276 in. (23 ft).

Basis: A design requirement will limit lift heights for HLW canisters to less than 23 ft above the bottom of a cask, waste package, staging rack, or staging pit (BSC 2004k, Section 5.1.1.24).

This assumption is used in Section 6.2.1.3.

4.20 The maximally exposed individual at the site boundary is assumed to receive doses from the inhalation, resuspension inhalation, air submersion, groundshine, and ingestion pathways for a period of 8,760 hr. The onsite individual member of the public, at 100 m or 3 km away from a DTF, the FHF, or a subsurface exhaust shaft is assumed to receive doses from inhalation, resuspension inhalation, air submersion, and groundshine pathways for a period of 2,000 hr.

Basis: It is expected that onsite ground contamination after event sequences will be detected and removed prior to restart of the repository operation. Unlike facility workers who work 2,000 hr a year, an individual member of the public is expected to visit the Yucca Mountain site for only a few days in a year. Therefore, the assumption of a 2,000-hr exposure period is conservative. One hundred meters is a conservative distance from the surface facility stack to any location outside the restricted area.

This assumption is used in Sections 6.2.1.1.3, 6.2.1.5, 6.3.1, and 6.3.3.

4.21 It is conservatively assumed that radionuclides are released from a height of 30 m during surface facility normal operations. This release height is used to calculate χ/Q values for use in dose calculations for surface facility normal operations.

Basis: A release height of 30 m is smaller than the stack heights of 40 m for DTF 1, of 40 m for DTF 2, or of 36 m for the FHF (BSC 2004f, Tables I-1, I-2, and Table I-14). A lower release height would result in a higher radiation dose. Hence, a release height of 30 m is conservative.

This assumption is used in Section 6.2.1.1.

4.22 For members of the public at the site boundary, the period of long-term exposure to ground contamination and intake of contaminated food is 1 year.

Basis: The assumption of 1-year exposure to ground contamination and intake of contaminated food is conservative when compared with an exposure period of 30 days recommended in ISG-5 (NRC 2003a, p. 9-15).

This assumption is used in Sections 3.1.2, 3.2.2, 6.1.4, 6.1.5, and 6.2.1.5.

4.23 It is assumed that 154 failed fuel rods per year are expected to be vulnerable to oxidation and cladding unzipping.

Basis: The number of failed SFAs expected to be received at the repository is estimated to be approximately 4 percent of the total, using data from NUREG/CR-3950 (Bailey and Wu 1990, Table 30), *Spent Nuclear Fuel Discharges from U.S. Reactors 1994* (DOE 1996, Table 5), and "Meeting the Challenge of Managing Nuclear Fuel in a Competitive Environment" (Yang 1997, Table 1), supplemented by data for the period 1980 to 2002 from *The Technical Basis for the Classification of Failed Fuel in the Back-End of the Fuel Cycle* (EPRI 1997, p. 4-1). The total number of SFAs expected is approximately 220,000 (BSC 2003c, Table 2). Using a repository operation period of 25 years (BSC 2004n, Table 1), 350 SFAs/year are expected to contain failed or damaged rods.

The majority of the failures are of a pinhole or hairline crack variety, while a range of 10 to 20 percent are failures of cladding larger than a pinhole or hairline crack, as indicated in EPRI (1997, p. 4-1). Therefore, about 70 SFAs/year are expected to be vulnerable to oxidation and cladding unzipping during fuel handling operations. On average, there are 2.2 rods per failed assembly (EPRI 1997, p. 4-1), which gives 154 failed fuel rods per year (= 70 SFAs × 2.2 rods/assembly).

This assumption is used in Section 6.2.1.3.

4.24 An ARF of 1.2×10^{-4} is assumed for fuel rods expected to be vulnerable to oxidation and cladding unzipping.

Basis: At temperatures of 500°C to 700°C, bare fuel pellets are fully oxidized to U_3O_8 in 1 hr according to *Accident Analysis for Continued Storage* (Davis et al. 1998, p. 9). The particle size distribution of the resultant powder was determined and the fraction of particles potentially small enough to become airborne was found to be 12 percent. Of this amount, approximately 1 percent is expected to be respirable. Therefore, the total ARF is 1.2×10^{-3}. This is consistent with the ARF of 1.0×10^{-3} determined by DOE (1994, p. 4-3) for complete oxidation of uranium metal in a flowing air atmosphere. Data from *Oxidation of Spent Fuel Between 250 and 360°C* (EPRI 1986) indicates that even after crack propagation (unzipping) begins, the rate of growth of the crack is slow. For example, the rate of growth at 360°C was determined to be 2.3×10^{-3} cm/min for a defect size of 760 μm in diameter (EPRI 1986, Table 3-3). At this rate, after 100 hr a 14 cm crack in the fuel rod would appear. This is approximately 4 percent of the total fuel rod length. Therefore, after 100 hr only 4 percent of the total fuel in the rod would be oxidized. Conservatively rounding up to 10 percent and using the ARF of 1.2×10^{-3} for bare fuel pellets, an ARF of 1.2×10^{-4} is assumed for fuel rods expected to be vulnerable to oxidation and cladding unzipping.

This assumption is used in Section 6.2.1.3.

4.25 Little or no oxidation is expected to occur for intact fuel or fuel with pinhole or hairline cracks during fuel handling operations in the repository. The fuel fine release fraction of 3×10^{-5} for a burst rupture is conservatively used for fuel with pinhole leaks or hairline cracks where little or no fuel oxidation occurs.

Basis: EPRI (1986, p. iii) determined that both the size and shape of the defect appear to influence the time to cladding splitting. For temperatures above 283°C, the time to cladding splitting was longer for the sharp small defect than for the large circular defect. This effect diminished as the temperature decreased. The breaches were induced by pressurizing the sample rods at elevated temperatures. Breach sizes ranged from 8 to 52 μm.

These breaches are usually axial cracks with pinhole protuberances through the outer cladding surface. The large circular defect was a hole of 760 μm drilled in the fuel rod (EPRI 1986, p. iii). For an 8 μm defect, it was found that the incubation time (defined as the time when a through-the-wall crack starts to propagate) was 455 hr as opposed to 79 hr for a 760 μm defect at 325°C. The incubation time for a 27 μm defect at 360°C was found to be between 52 hr and 60 hr while the incubation time for a 760 μm defect at the same temperature was 20 hr (EPRI 1986, p. iii). Assuming that the breaches of 8 to 52 μm represent pinhole leaks or hairline cracks, the minimum incubation time is 52 hr.

The tests discussed in this assumption were performed for fuel with a burnup of ~27 GWd/MTU (EPRI 1986, Table 2-2). The average expected burnup for fuel received at the repository is 48 GWd/MTU for PWR fuel (Table 3) and 40 GWd/MTU for BWR fuel (Table 3). The plateau duration, which is a component of the incubation time, is strongly dependent on burnup and temperature, as indicated by *The Burnup Dependence of Light Water Reactor Spent Fuel Oxidation* (Hanson 1998, Figure 5.6). For fuel oxidized at 305°C and a burnup of 27 GWd/MTU the plateau duration is approximately 13 hr, for a burnup of 40 GWd/MTU the plateau duration is approximately 440 hr, and for a burnup of 48 GWd/MTU the plateau duration is over 1,000 hr (Hanson 1998, Figure 5.6).

NUREG/CR-0722 (Lorenz et al. 1980, p. 34) states that releases of gases from a fuel rod cease after 4 hr of testing of a high burnup fuel at 500°C in dry air, suggesting that the defect hole became plugged and prevented further oxidation of the fuel.

This assumption is used in Section 6.2.1.3.

5. USE OF COMPUTER SOFTWARE

5.1 SOFTWARE APPROVED FOR QUALITY-AFFECTING WORK

MACCS2 (ORNL 1998) is the only computer program used in this calculation except for the Microsoft Excel 97 spreadsheet program. MACCS2 (ORNL 1998) is baselined in accordance with LP-SI.11Q, *Software Management*. The code validation and verification results are documented in *Users Manual: MACCS2 Version 1.12, STN: 11000-1.12-00* (Tsai 2003).

<u>MACCS2 V.1.12 (ORNL 1998)</u>

- Program Name: MACCS2
- Version/Revision Number: Version 1.12
- Software Tracking Number (STN): 11000-1.12-00
- Computer Type: Dell OPTIPLEX GX240
- Platform/Operating System: PC/Windows 2000
- Computer Processing Unit Tag Number: CRWMS M&O Tag 150446.

MACCS2 (ORNL 1998) calculates doses resulting from accidental radionuclide releases from nuclear facilities to onsite and offsite members of the public. The code considers: atmospheric transport; short- and long-term mitigation actions; and potential exposure pathways, such as inhalation, resuspension inhalation, ingestion, air submersion, and groundshine.

MACCS2 (ORNL 1998) simulates the impact of accidental atmospheric releases of radiological materials on the surrounding environment. The principal phenomena considered in MACCS2 (ORNL 1998) are atmospheric transport, mitigative actions based on dose projection, dose accumulation by a number of pathways including food and water ingestion, and acute and chronic exposures. MACCS2 (ORNL 1998) contains simple models with analytical solutions. A MACCS2 (ORNL 1998) calculation consists of three phases: input processing and validation, phenomenological modeling, and output processing. The phenomenological models are based on empirical data and the solutions are analytical in nature and computationally straightforward.

The modeling phase is divided into three modules: ATMOS, EARLY, and CHRONC. The ATMOS module treats atmospheric transport and dispersion of material and its deposition from the air using a Gaussian plume model with Pasquill-Gifford dispersion parameters. The EARLY module models consequences of the accident to the surrounding area during an emergency action period. The CHRONC module considers the long-term impact in the period subsequent to the emergency action period.

The applicability of MACCS2 (ORNL 1998) to normal operations is discussed in Section 3.1.2. The MACCS2 (ORNL 1998) calculated χ/Q values (Attachment C) are more conservative than the χ/Q values calculated based on Regulatory Guide 1.111 for normal operations (Section 3.1.2).

5.2 EXEMPT SOFTWARE

The Microsoft Excel 97 spreadsheet program is used to perform simple calculations as documented in Section 6. User-defined formulas, input, and results are documented in sufficient detail in Section 6 to allow for independent duplication of various computations without recourse to the originator. This software is considered exempt from the requirements of LP-SI.11Q, *Software Management.*

6. ANALYSIS

6.1 RADIONUCLIDE RELEASES FROM NORMAL OPERATIONS AND EVENT SEQUENCES

This section discusses radionuclide releases from normal operations and Category 1 and Category 2 event sequences, as well as event sequence frequency calculation. Waste forms involved in normal operations and in Category 1 and Category 2 event sequences include PWR or BWR SFAs, HLW, and naval SNF (Section 4, Assumption 4.1).

6.1.1 Repository Operations

NUREG-1804 (NRC 2003b, p. 2.1-29) requires a discussion of modes of geologic repository operations area operation. This section discusses facility operations in the repository operations area, including the CHF, DTF 1, DTF 2, the FHF, the TCRRF, the subsurface facility, and the SNF aging system. Operations relevant to consequence analyses may include waste handling, maintenance, waste retrieval, and SNF aging.

Normal operations associated with waste handling, waste retrieval, and SNF aging are discussed in Sections 6.1.2.1 through 6.1.2.3. Doses to facility workers from general facility maintenance operations are discussed in Section 6.4.1.2.

6.1.2 Radionuclide Releases from Normal Operations

This section discusses normal operations in surface and subsurface facilities and the potential release of radionuclides from these facilities.

6.1.2.1 Potential Releases from Surface Facilities

The repository surface facilities are located near the North Portal entrance to the repository. Waste handling operations are carried out in the geologic repository operations area. The facilities necessary to receive, package, age, and emplace waste in the repository are located in this area.

Waste transfer operations are performed in several waste handling facilities, including the CHF, DTF 1, DTF 2, and the FHF, as described in *Internal Hazards Analysis for License Application* (BSC 2004q, Section 6.4).

The SNF and vitrified HLW are transported to the repository in NRC-certified transportation casks in accordance with applicable federal regulations. When an SNF or HLW transportation cask is received, personnel verify the shipping manifests and inspect and survey the cask and its trailer or railcar. The cask is then moved to a restricted area for staging (BSC 2004q, Section 6.4).

From the staging area, casks are delivered to TCRRF for transfer to a site rail transfer cart and the transportation cask buffer area, as applicable. Casks delivered to the FHF are delivered directly to the facility on public conveyance (truck trailer or railcar) from the staging area; there is no FHF interface with a site rail transfer cart or transportation cask buffer area (BSC 2004q, Section 6.4).

When an empty waste package is scheduled for processing, it is moved into the appropriate waste handling facility. The waste handling facility operators prepare and configure the empty waste package for SNF/HLW transfer. The steps for waste package preparation include positioning the waste package for transfer, inserting the inner shell lid and transferring the waste package to the appropriate SNF/HLW transfer area (BSC 2004q, Section 6.4).

When a waste shipment is scheduled for processing, the transportation cask containing the SNF/HLW is moved into an appropriate waste handling facility. Cask preparation consists of (BSC 2004q, Section 6.4):

- Surveying the cask for radiation and surface contamination
- Decontamination, if required
- Removal of impact limiters
- Upending the cask
- Positioning the cask for transfer
- Sampling gases inside the cask
- Venting the cask cavity to atmospheric pressure
- Removing the cask lid(s)
- Transferring the cask to the SNF/HLW transfer area
- Establishing radiological confinement between the cask and the waste handling facility, as required.

Radiological confinement is attained using the cask unload port sealing device, if required. If the cask shipment contains a dual-purpose canister (DPC) of commercial SNF, then the DPC undergoes a series of preparation steps prior to commercial SNF transfer. These preparation steps include the rigging of the DPC for transfer, removing the DPC to the SNF/HLW transfer area, sampling gases inside the DPC, cutting the DPC lid off, and removing the DPC lid to allow for commercial SNF transfer. The empty DPC and lid are packaged as low-level radioactive waste and transported to a low-level radioactive waste disposal site (BSC 2004q, Section 6.4).

If a waste shipment contains high-heat commercial SNF, then thermal aging of the waste form may be required. In this event, commercial SNF is transferred to site specific casks for transport to the SNF aging system pads. When a site specific cask is scheduled for processing, it is moved into an appropriate waste handling facility. The waste handling facility operators inspect, prepare, and configure the site specific cask for SNF/HLW transfer.

Preparation of a site specific cask consists of positioning the cask for transfer, removing the cask lid(s), transferring the cask to the SNF/HLW transfer area, and establishing radiological confinement between the empty cask and the waste handling facility, if required. The SNF/HLW is remotely transferred from the transportation casks to the site specific cask. Once loaded, the site specific cask is closed, inerted (if required), and sealed prior to being transported to the SNF aging system for cooling. In addition, if staging is required, then SNF/HLW may be placed in a site specific cask and transported to the SNF aging system (BSC 2004q, Section 6.4). No normal operational releases are expected from the SNF aging system as reported in *Aging-Related Dose Assessment* (Migliore 2004).

After waste package loading is completed, the inner lid is placed on the waste package. Then the waste package is transferred to the waste package closure cell where the inner lid is welded first and then the middle and outer lids are placed on the waste package and welded, the welds are inspected, and the weld stresses are mitigated. Waste packages may be staged (DTF 1 and DTF 2 only) or transferred to the waste package loadout area where the waste package is surveyed for contamination, down-ended onto the emplacement pallet, positioned for lifting collar removal, and transferred into the waste package transporter. The waste package transporter is moved into the subsurface facilities to an emplacement drift, where the waste package is remotely transferred to an emplacement gantry for final positioning in the emplacement drift (BSC 2004q, Section 6.4).

It is assumed that 3,000 MTHM/year (Section 4, Assumption 4.2) of commercial SNF pass through DTF 1, DTF 2, and the FHF.

It is further assumed that 1 percent (Section 4, Assumption 4.3) of the fuel rods in the nominal annual throughput have cladding breaches and that fission product gases, volatile species, and fuel fines are released to the environment. Therefore, releases from 1 percent of fuel rods in the 6,316 PWR SFAs are used as the source term for the calculation of normal operations doses.

It is assumed that during normal operations of the surface facilities, there are releases from the exhaust stacks of DTF 1, DTF 2, and the FHF (Section 4, Assumption 4.17). There are no releases expected from the CHF during normal operations, because the CHF handles only sealed canisters and a sealed canister does not leak.

Failed fuel is defined in 10 CFR Part 961, p. 607, as the fuel that has failed cladding, namely, hairline cracks and pinhole leaks or larger. Failed fuel has already released its radionuclide inventory in the gap region between the cladding and the fuel pellets, while at a nuclear power plant. Historical fuel discharge data show that the fuel rod failure rate is about 0.01 percent, versus a much smaller range of 0.02 to 0.07 percent in the first 20 years of commercial nuclear power (BSC 2003c, p. G-6) (Section 4, Assumption 4.3).

Failed commercial SNF will be shipped to the repository in screen-end or closed-end canisters in a transportation cask. Failed commercial SNF canisters will be transferred from the cask to a waste package inside a waste transfer cell in a DTF or the FHF, during normal operations. Exposing the failed fuel in air while in a DTF or the FHF, could cause oxidation of the fuel (UO_2) to higher oxides, such as U_3O_8. When U_3O_8 starts to form, the fuel pellet volume expands and eventually cladding could unzip.

It is assumed that for normal operations, radioactive materials are released in a duration of 24 hr (Section 4, Assumption 4.6) via the building exhaust stack (Section 4, Assumption 4.7) as a radioactive plume that is dispersed en route to the site boundary. The release is assumed to result in an acute individual exposure during plume passage and a chronic individual exposure to ground contamination and contaminated food after plume passage (Section 4, Assumption 4.6). The period of long-term exposure to ground contamination and intake of contaminated food is assumed to be 1 year (Section 4, Assumption 4.6).

It is assumed that for public and worker dose calculations, the HVAC system is operating and no airborne material leaks into space occupied by workers in rooms adjacent to a waste transfer cell in a DTF or the FHF (Section 4, Assumption 4.13). This assumption considers that the waste transfer cell confinement and the HVAC system are designed to prevent leakage to rooms adjacent to the transfer cell in the event of a Category 1 event sequence.

Credit is taken for HEPA filters to remove radionuclides.

6.1.2.2 Potential Releases from Subsurface Facility

Under normal operations of the subsurface repository, there are three potential mechanisms that generate airborne releases of radioactive materials, which are:

- Resuspension of radioactive contamination from the external surfaces of the emplaced waste packages
- Neutron activation of ventilating air inside the emplacement drifts
- Neutron activation of silica dust generated from host rock in the emplacement drifts.

Contamination on the surface of waste packages in emplacement drifts may become entrained in the ventilated airflow and released to the environment through the ventilation shaft. Neutron flux, that penetrates the emplaced waste packages during normal operations, activates ventilated airflow through subsurface emplacement drifts and through the host rock surrounding the emplaced waste packages. Activated air and dust are released to the environment through the ventilation shaft. Total estimated releases from the subsurface facility from normal operations are shown in Table 8 (BSC 2004g, Table 14).

Table 8. Annual Releases from Subsurface Facility Normal Operation

Normal Operations Release (Ci/year)	
Waste Package Surface Contamination	
Cs-137	6.8E-03
Crud Co-60	2.9E-03
Co-60	7.8E-06
Ni-63	6.3E-06
Sr-90	6.8E-04
Y-90	6.8E-04
Pm-147	3.0E-06
Sm-151	5.3E-06
Eu-154	1.7E-05
Pu-241	6.2E-04
Pu-238	5.7E-05
Pu-239	4.4E-06
Pu-240	7.9E-06
Am-241	4.9E-05
Am-243	5.5E-07

Table 8. Annual Releases from Subsurface Facility Normal Operation (Continued)

Normal Operations Release (Ci/year)	
Waste Package Surface Contamination (Continued)	
Cm-243	2.6E-07
Cm-244	3.4E-05
Waste Package Surface Contamination	
Activated Air	
N-16	3.4E-02
Ar-41	2.0E+01
Activated Dust	
N-16	9.5E-13
Na-24	3.7E-03
Al-28	1.6E-03
Si-31	5.2E-04
K-42	8.0E-04
Fe-55	8.2E-05

Source: BSC 2004g, Table 14

Annual releases (Table 8) are based on the postulated activation of air and silica dust in the subsurface facilities during normal operations as summarized in BSC (2004g, Table 14). Subsurface releases include waste package surface contamination and radionuclides generated by the activation of air (^{41}Ar and ^{16}N) and dust (^{16}N, ^{24}Na, ^{28}Al, ^{31}Si, ^{42}K and ^{55}Fe). Nitrogen-16 (^{16}N) is not considered in the dose assessment because of its short half-life and the associated long travel time from the release point to the site boundary.

The releases from the subsurface exhaust shafts during normal operations are assumed to be at ground level (Section 4, Assumption 4.17).

6.1.2.3 Potential Releases from Retrieval Operations

The geologic repository operations area must be designed to preserve the option of waste retrieval throughout the period during which wastes are emplaced and thereafter, until NRC review of the information obtained from a performance confirmation program is completed (10 CFR 63.111(e)(1)). A permanent retrieval could potentially require the entire underground repository to be emptied. For permanent retrieval, a separate license would be required to build new facilities. No airborne releases of radionuclides are expected for normal retrieval operations.

6.1.3 Event Sequence Frequency Calculation and Categorization

Categorization of event sequences based on frequency calculations and design bases are presented in BSC (2004k, Section 7). Design bases are a set of requirements to be implemented by design to prevent releases or to mitigate radiological consequences. Bounding Category 1 and Category 2 event sequences are identified in BSC (2004k, Section 7).

6.1.3.1 External Event Sequences

If there is some chance that an external hazard could initiate an event sequence, then the hazard is considered applicable. For example, tsunami is not applicable because it cannot happen at the repository site, but aircraft crash is applicable because it could happen at the site and an aircraft crash is not easily screened out. In some cases, event sequences initiated by applicable hazards were shown to be beyond Category 2 by separate calculations.

The following hazards are applicable to the repository (BSC 2004k, Section 6.2):

- Aircraft crash
- Ash fall due to volcanism
- Drift degradation
- Extreme weather (temperature) fluctuation
- Lightning
- Loss of offsite or onsite power
- Nearby industrial and military activities
- Rainstorm and flooding
- Range fire
- Sandstorm
- Seismic activity
- Tornado and extreme wind.

Category 1 or Category 2 event sequences initiated by external hazards are prevented by designing SSCs important to safety or engineered features to withstand external hazards or by showing that the event frequency is beyond Category 2 (BSC 2004k, Section 6.2). Implementation of seismic design requirements for SSCs that are important to safety ensures that surface facilities are designed such that seismic events will not initiate Category 1 or Category 2 event sequences that exceed the dose limits specified in 10 CFR Part 63 (BSC 2004k, Section 4.1.16).

6.1.3.2 Internal Event Sequences

Internal events involving fires are screened out (BSC 2004k, Section 4.1.9) by design and operational requirements that limit the quantities of combustible or flammable materials that are present in the areas where SNF is processed or staged (BSC 2004k, Section 5.1.1.6).

Internal events involving criticality are prevented by design requirements that control the amounts of moderators in the CHF, DTF 1, DTF 2, and the FHF. In addition, in the design of SNF staging racks and the remediation pool, geometry control is used (BSC 2004k, Sections 5.1.1.2 and 5.1.5.9).

An identification and categorization of internal event sequences relating to retrieval operations is beyond the scope of this analysis. Permanent retrieval activities can be examined in future preclosure safety analysis to identify potential event sequences.

Certain internal events involving drops or collisions of commercial SNF assemblies are Category 1 events. The frequencies of drops or collisions and the numbers of SFAs affected at the maximum rate of receipt are presented in Table 9.

Table 9. Category 1 Event Sequences

Event Sequence Identifier	Description	Event Frequency for Consequence Analyses (events/year)	MAR
GET-03D	Drop of a commercial SNF assembly	0.5	Two PWR or two BWR SFAs
GET-03B	Collision involving a commercial SNF assembly	0.5	One PWR or one BWR SFA

BWR = boiling water reactor; GET = generalized event tree; MAR = material at risk; PWR = pressurized water reactor; SNF = spent nuclear fuel.

Source: BSC 2004k, Section 7.2.1

Drops or collisions involving uncanistered commercial SFAs are determined to be Category 1 event sequences (BSC 2004k, Table 28), because of the large number of commercial SFAs to be handled inside a waste transfer cell of a DTF in 1 year. The frequency of drops or collisions of uncanistered commercial SNF during dry transfer depends on the handling frequency and the drop or collision rate per movement.

6.1.4 Radionuclide Releases from Category 1 Event Sequences

6.1.4.1 Event Sequence GET-03D (Table 9)

During the transfer of an individual commercial SFA from a transportation cask to a waste package in a transfer cell, an SFA could drop:

- Into an empty cask, waste package, or transfer cell staging rack
- Onto another SFA in the cask
- Into a waste package
- Onto the transfer cell floor.

It is assumed that the cladding of 100 percent of the fuel rods is damaged and that radionuclides are released to the environment (Section 4, Assumption 4.10). The release of radionuclides is divided into two phases. The first phase involves a release of fission product gases, volatile species, fuel fines, and crud immediately following a cladding breach. The second phase involves oxidation of UO_2 to U_3O_8 and cladding unzipping. During the fuel oxidation period, only fission product gases and volatile species are released. The release fractions for the first phase are given in Table 5. The release fractions of fission product gases and volatile species from fuel oxidation are included in the release fractions given in Table 5.

It is assumed that for this event sequence, radioactive materials are released in a duration of 1 hr (Section 4, Assumption 4.5) as a ground-level plume (Section 4, Assumptions 4.5 and 4.17) that is dispersed en route to the site boundary, resulting in an acute individual exposure during plume passage and a chronic individual exposure to ground contamination and contaminated food after plume passage (Section 4, Assumption 4.5). The period of long-term exposure to ground contamination and intake of contaminated food is 1 year (Section 4, Assumption 4.5).

It is assumed that for worker dose calculations, the HVAC system is operating and no airborne material released from Category 1 event sequences leaks into space occupied by workers in rooms adjacent to the waste transfer cell of a DTF or the FHF (Section 4, Assumption 4.13), because the transfer cell confinement and the HVAC system are designed to prevent any leakage to rooms adjacent to the transfer cell in the event of a Category 1 event sequence.

Credit is taken for HEPA filters to remove radionuclides (Section 4, Assumption 4.4).

6.1.4.2 Event Sequence GET-03B (Table 9)

During the transfer of individual commercial SFAs from a transportation cask to a waste package in a transfer cell, an SFA could collide with heavy equipment or structures. It is assumed that the cladding of 100 percent of the fuel rods is damaged and that radionuclides are released to the environment (Section 4, Assumption 4.10).

The release of radionuclides is divided into two phases. The first phase involves a release of fission product gases, volatile species, fuel fines, and crud immediately following a cladding breach. The second phase involves oxidation of UO_2 to U_3O_8 and cladding unzipping. During the fuel oxidation period, only fission product gases and volatile species are released. The release fractions for the first phase are given in Table 5. The release fractions of fission product gases and volatile species from fuel oxidation are included in the release fractions given in Table 5.

It is assumed that for this event sequence, radioactive materials are released in a duration of 1 hr (Section 4, Assumption 4.5) as a ground-level plume (Section 4, Assumptions 4.5 and 4.17) that is dispersed en route to the site boundary, resulting in an acute individual exposure during plume passage and a chronic individual exposure to ground contamination and contaminated food after plume passage (Section 4, Assumption 4.5). The period of long-term exposure to ground contamination and intake of contaminated food is 1 year (Section 4, Assumption 4.5).

It is assumed that for worker dose calculations, the HVAC system is operating and no airborne material released from Category 1 event sequences leaks into space occupied by workers in rooms adjacent to the waste transfer cell in a DTF or the FHF (Section 4, Assumption 4.13), because the transfer cell confinement and the HVAC system are designed to prevent any leakage to rooms adjacent to the transfer cell in the event of a Category 1 event sequence.

Credit is taken for HEPA filters to remove radionuclides (Section 4, Assumption 4.4).

6.1.5 Radionuclide Releases from Category 2 Event Sequences

Drops or collisions of SFAs, HLW, and naval SNF in canisters, transportation casks, or waste packages are determined to be Category 2 event sequences (BSC 2004k, Section 7). It is assumed that the cladding of spent fuel rods is damaged and that radionuclides are released in a 1-hr duration as a ground-level release (Section 4, Assumption 4.5). Two Category 2 event sequences, that bound other Category 2 event sequences (BSC 2004k, Section 7) with respect to frequency and MAR, are identified in Table 10.

Table 10. Bounding Category 2 Event Sequences

Event Sequence Identifier	Description	Expected Number of Occurrences Over the Life of the Repository	Bounding Material at Risk
GET-01A	Drop of a transportation cask, without impact limiters, in the CHF, DTF, FHF, or TCRRF	5.8E-01	74 BWR or 36 PWR SFAs, 5 DOE HLW canisters, or 1 naval SNF canister
GET-02B	Drop of the inner lid of a transportation cask, site specific cask, or waste package into a transportation cask, site specific cask, or waste package in the CHF, DTF, or FHF	5.8E-01	74 BWR or 36 PWR SFAs, 5 DOE HLW canisters or 1 naval SNF canister

BWR = boiling water reactor; CHF = Canister Handling Facility; DOE = U.S. Department of Energy; DTF = Dry Transfer Facility (DTF 1 or DTF 2); FHF = Fuel Handling Facility; HLW = high-level radioactive waste; PWR = pressurized water reactor; SNF = spent nuclear fuel; TCRRF = Transportation Cask Receipt and Return Facility.

Source: BSC 2004k, Table 59

6.1.5.1 Bounding Category 2 Event Sequence Involving Boiling Water Reactor and Pressurized Water Reactor Spent Nuclear Fuels

Events GET-01A and GET-02B (Table 10) bound other Category 2 event sequences involving PWR and BWR SNFs. Event GET-01A involves a drop of a transportation cask and breach of PWR or BWR SFAs. It is assumed that the event sequence results in breaches of 36 PWR SFAs or 74 BWR SFAs and releases of radionuclides to the environment.

It is assumed that for this event sequence, radioactive materials are released in a duration of 1 hr as a ground-level radioactive plume that is dispersed en route to the site boundary, resulting in an acute individual exposure during plume passage and a chronic individual exposure to ground contamination and contaminated food after plume passage (Section 4, Assumption 4.5). The period of long-term exposure to ground contamination and intake of contaminated food is 1 year (Section 4, Assumption 4.5).

No credit is taken for HEPA filters to remove radionuclides (Section 4, Assumption 4.4).

6.1.5.2 Bounding Category 2 Event Sequence Involving Naval Spent Nuclear Fuel

Events GET-01A and GET-02B (Table 10) bound other event sequences involving naval SNF. Event GET-01A (Table 10) involves a drop of a transportation cask and breach of a naval SNF canister. It is assumed that the event sequence results in damage to naval SNF assemblies and releases of radionuclides to the environment.

It is assumed that for this event sequence, radioactive materials are released in a duration of 1 hr as a ground-level radioactive plume that is dispersed en route to the site boundary, resulting in an acute individual exposure during plume passage and a chronic individual exposure to ground contamination and contaminated food after plume passage (Section 4, Assumption 4.5). The period of long-term exposure to ground contamination and intake of contaminated food is 1 year (Section 4, Assumption 4.5).

No credit is taken for HEPA filters to remove radionuclides (Section 4, Assumption 4.4).

6.1.5.3 Bounding Category 2 Event Sequence Involving Vitrified HLW

Events GET-01A and GET-02B (Table 10) bound other event sequences involving vitrified HLW. Event GET-01A (Table 10) involves a drop of a transportation cask and breach of five vitrified HLW canisters. It is assumed that the event sequence results in breaches of five HLW canisters and releases of radionuclides to the environment.

It is assumed that for this event sequence, radioactive materials are released in a duration of 1 hr as a ground-level radioactive plume that is dispersed en route to the site boundary, resulting in an acute individual exposure during plume passage and a chronic individual exposure to ground contamination and contaminated food after plume passage (Section 4, Assumption 4.5). The period of long-term exposure to ground contamination and intake of contaminated food is 1 year (Section 4, Assumption 4.5).

No credit is taken for HEPA filters to remove radionuclides (Section 4, Assumption 4.4).

6.1.6 Interactions of Hazards and Controls

Much of the equipment used for handling and moving SNF and HLW, such as cranes, gantries, and transporters, is remotely operated. To prevent the initiation of a Category 1 or Category 2 event sequence upon a loss of power, the instrumentation and control systems for remotely operated equipment are designed to be failed safe so that a crane or gantry does not drop loads (BSC 2004k, Section 5.1.2.1).

There are no procedural safety controls, such as operator actions, credited to prevent a Category 1 or Category 2 event sequence (BSC 2004k, Section 5.1.2.1).

The HVAC system is required to mitigate potential releases of radionuclides from surface facilities during normal operations and Category 1 event sequences. This system is required to be operable when waste handling operations are conducted in a surface waste handling facility. The fan and HEPA filters are not required to be operable following a Category 2 event sequence.

Facility design features, such as moderator control in the waste transfer cell, are required to prevent criticality.

6.2 INPUTS

This section discusses the input parameters used in public and worker dose calculations.

6.2.1 Public Dose Calculation Design Inputs

This section discusses input files, output files, and input parameters used in MACCS2 (ORNL 1998) analyses of radionuclide releases from normal operations and Category 1 and Category 2 event sequences.

6.2.1.1 MACCS2 Input Files

Seven input files are necessary to run MACCS2 (ORNL 1998) for event sequences. The first file, ATMOS, provides radionuclide inventory and release information. The second file, EARLY, provides information related to the early phase of the event sequence. The third file, CHRONC, provides information on the long-term chronic exposure phase. A fourth file provides hourly meteorological data, and a fifth file provides site-specific data including population data, land use data, and grid spacing information. The remaining two input files, those being the dosimetry file and the food pathway file, are identified in the first five input files. The dosimetry file, current applications use YMPD825.INP, is defined in the EARLY input file YMPEA1.INP. The food pathway file, current applications use YUCCALA.BIN, is defined in the CHRONC input file YMPCH1.INP. These file names are not changed; if the dosimetry file is changed, then it is necessary to rerun the COMIDA2 module with the same dosimetry file because MACCS2 checks the two files to ensure that they are based on the same dosimetry file used in the current MACCS2 run.

MACCS2 (ORNL 1998) is run using a batch command file called RUNMAX2. Use of this file requires specification of additional file names that identify the files' source of input information required by MACCS2.

The command line for running the PWR base case is:

RUNMAX2 LAPWR01 YMPEA1 YMPCH1 YMP1999 YMPSIT LAPWR01

This command line directs MACCS2 (ORNL 1998) to read the ATMOS data set from file LAPWR01.INP, the EARLY data set from YMPEA1, the CHRONC data set from YMPCH1, the meteorology data set from YMP1999, and the site data set from file YMPSIT. The output results are written to file LAPWR01.OUT. The dosimetry file YMPD825.INP and the food pathway file YUCCALA.BIN are used as previously described.

The command line for running the HLW base case is:

RUNMAX2 LAHLW01 YMPEA YMPCH1 YMP1999 YMPSIT LAHLW01

This command line directs MACCS2 (ORNL 1998) to read the ATMOS data set from file YMHLW01.INP, the EARLY data set from YMPEA, the CHRONC data set from YMPCH1, the meteorology data set from YMP1999, and the site data set from file YMPSIT. The output results are written to file LAHLW01.OUT.

The command line for running the naval SNF base case is:

RUNMAX2 LANAV01 YMPEA YMPCH1 YMP1999 YMPSIT LANAV01

This command line directs MACCS2 (ORNL 1998) to read the ATMOS data set from file LANAV01.INP, the EARLY data set from YMPEA, the CHRONC data set from YMPCH1, the meteorology data set from YMP1999, and the site data set from file YMPSIT. The output results are written to file LANAV01.OUT.

To modify one of the base cases for a different event sequence, the following parameters are defined and entered into the ATMOS file. The new file is prepared by first copying the appropriate existing file, for example LAPWR01.INP, to a file with a representative name, such as changing the "01" to a "02" to represent a new event sequence and then editing the file to represent the new event sequence.

Six sets of ATMOS input files are prepared based on different fuel types: a Maximum PWR SFA of 5 years decay, an Average PWR SFA of 25 years decay, a Maximum BWR SFA of 5 years decay, an Average BWR SFA of 25 years decay, a HLW canister, and a naval fuel canister. These files are used to describe additional event sequences by modifying specific parameters in the input files.

The parameters likely to need modification for a new event sequence are listed in Table 11.

Each parameter listed in Table 11 is entered in the ATMOS type file using a keyword. Table 11 also provides the keyword and section in the MACCS2 user manual (ORNL 1998) that describe the parameter. Base case files are set up to release the activity over a 1-hr period. If a different release period is desired, then that value is entered. If a different value is entered, then the value for parameter PMTIMBAS is also changed to the same value. This inhibits special plume meander corrections. The maximum value allowed for the release duration is 24 hr.

The values for the initial sigma-y and sigma-z values should not be changed unless the building height or width (not an input) is changed. An initial sigma-y of 10.3 m and an initial sigma-z of 12 m (Table 11) are for a 25.9-m high building that is 44.5 m wide (i.e., FHF building dimensions) (BSC 2004f, Table 2) and a release height of 0 m (ground level) (Section 4, Assumption 4.17) or 30 m (Section 4, Assumption 4.21). These values are used for each MACCS2 run except for Run 3 (Section 6.3). An initial sigma-y of 0.1 m and an initial sigma-z of 0.1 m (Table 11) are used for MACCS2 Run 3 (Section 6.3) in which no building structures are present at the subsurface exhaust shafts.

Table 11. MACCS2 Key Input Parameters

Parameter Description	Keyword	ORNL 1998 Section	Base Case Value
Release height (m)	RDPLHITE	5-26	0., 30
Building height (m)	WEBUILDH	5-22	25.9
Release duration (s)	RDPLUDUR	5-26	3600, 86400
Heat of release (watts)	RDPLHEAT	5-25	0.
Initial sigma-y value (m)[a]	SIGYINIT	5-22	0.1, 10.3
Initial sigma-z value (m)[a]	SIGZINIT	5-23	0.1, 12.0
Release fraction for H-3	RDRELFRC	5-28, 5-29	Table 5
Release fraction for Kr-85	RDRELFRC	5-28, 5-29	Table 5
Release fraction for Cs	RDRELFRC	5-28, 5-29	Table 5
Release fraction for Ru	RDRELFRC	5-28, 5-29	Table 5
Release fraction for Co	RDRELFRC	5-28, 5-29	Table 5
Release fraction for I	RDRELFRC	5-28, 5-29	Table 5
Release fraction for others	RDRELFRC	5-28, 5-29	Table 5

NOTE: [a] Values for the initial sigma-y and sigma-z values are conservative and have not changed.

Sources: ORNL 1998, Section 5, pp. 5-22, 5-23, 5-25, 5-26, 5-28, and 5-29;
Section 4, Assumptions 4.17 and 4.21.

If a different size building were involved, then both of the following would apply:

- The initial sigma-y value is evaluated as: building width / 4.3 m
- The initial value of sigma-z is evaluated as: building height / 2.15 m.

Release fractions are entered on one line in the order indicated. Release fractions are multiplied by the radionuclide inventory of one assembly or one canister to determine the total activity of each radionuclide released to the atmosphere in the event sequence. Release fractions are defined based on the core inventory, which represents the total activity in one SFA. If more than one SFA is involved in the event sequence, then the number of SFAs is included in the definition of the release fractions; if two SFAs are involved, then the release fractions are doubled.

6.2.1.1.1 MACCS2 EARLY and CHRONC Files

The MACCS2 (ORNL 1998) EARLY component file, YMPEA.INP, and the CHRONC component file, YMPCH1.INP, are not changed for consequence analysis. These files define the parameters for describing exposures during the early and late phases of an event sequence. The analyses are set up to eliminate evacuation so that people do not move, and to eliminate interdiction activities, such as impounding contaminated food, in the late phase, which results in an analysis that maximizes the exposure to the pathways. These files also allow the user to select the output results desired.

6.2.1.1.2 MACCS2 Meteorological Data File

Meteorology data from the Yucca Mountain site is available for the 5-year period of 1998 through 2002. The meteorological data used for χ/Q value calculations using the MACCS2 (ORNL 1998) code are based on site-specific measurements made at air quality and meteorology monitoring Site 1, which is a 60-m tower located approximately 1-km south–southwest of the North Portal.

The data consists of hourly data collected under quality assurance procedures that are submitted to the Technical Data Management System under the following data tracking numbers (DTNs):

DTN: MO0306WPMM9802.000. Wind Direction Sector, Wind Speed, Stability Class and Precipitation Data for MACCS2 98-02 Model.

DTN: MO0304MACCS280.000. Wind Direction Sector, Wind Speed, Stability Class and Precipitation Data for MACCS2 98-00 Model.

DTN: MO0210MXHT8491.000. Mean Seasonal Mixing Height Data (0400 and 1600 PST) for Desert Rock, Nevada from 1984 through 1991.

MACCS2 (ORNL 1998) requires one full year of hourly observation data. Five one-full-year hourly data files, 1998 through 2002, are located in DTN: MO0306WPMM9802.00. The remaining two DTNs were used as input sources to DTN: MO0306WPMM9802.00. Blanks in the files, indicating data gaps, are filled by directly inserting data or adjusting the data before it is inserted, from nearby Yucca Mountain Project meteorological stations, or by directly inserting data from different time periods having similar meteorological conditions at the primary station, Site 1.

Details on data replacement are documented in the data replacement methodologies accompanying the data input files in DTNs: MO0306WPMM9802.000 and MO0304MACCS280.000. Such data replacement was accomplished using professional judgement and specific knowledge of the meteorological conditions at Yucca Mountain.

The resulting meteorological data files are named:

- YMP1998.INP data for calendar year 1998
- YMP1999.INP data for calendar year 1999
- YMP2000.INP data for calendar year 2000
- YMP2001.INP data for calendar year 2001
- YMP2002.INP data for calendar year 2002.

The YMP1999.INP was selected for this calculation after a previous sensitivity analysis, "Met Data" (McDonnell 2003), showed that any of the five files produced similar results.

6.2.1.1.3 MACCS2 Output File

MACCS2 (ORNL 1998) output results are printed to the last named file on the command line; extension OUT. The first part of the file is a record of the parameter values used in the analysis and provides a record of the analysis. Results are provided as mean values and for 50th, 90th, 95th, 99th, and 99.5th percentiles. Three sets of results are provided: χ/Q values, whole body effective dose, and organ doses. Information on dose contributions from specific pathways is also contained in the output file.

A descriptive title is entered for the event sequence in the ATMOS file, near the start; keyword name RIATNAM1. The title is enclosed in single quote marks. The title appears on the output file as a check that the correct file was used in the analysis. A convention in the input files is that lines starting with an asterisk (*) are comment lines and are not read by the program. Only those lines recognized as containing data are given a number in the left column.

In the MACCS2 output file, the values of the parameter "Ground-Level Dilution, χ/Q (s/m^3)" at 100 m, 8 km, or 11 km for mean values and for 50th, 90th, 95th, 99th, and 99.5th percentiles are provided. Immediately following the χ/Q value outputs, the centerline doses at various distances away from the release point are provided for mean values and for 50th, 90th, 95th, 99th, and 99.5th percentiles. The centerline doses include the parameter "L-EFFECTIVE TOT LIF," for whole body dose excluding ingestion dose, and the parameter "L-BONE SUR TOT LIF," which is the highest TODE excluding ingestion dose. Among the internal organs, bone surface is the organ that receives the highest radiation dose. The ingestion dose is printed on the last page of the output file under the heading "MAXIMUM ANNUAL FOOD DOSE (EFFECTIVE)."

The offsite public TEDE, at 11 km or 8 km away from the release point for normal operations and Category 1 event sequences, is determined by summing up the mean value of the parameter "L-EFFECTIVE TOT LIF" and the ingestion dose. The TEDE at 11 km or 8 km from the release point for Category 2 event sequences is determined by summing up the 95th percentile value of the parameter "L-EFFECTIVE TOT LIF" and the ingestion dose. The TODE at 11 km or 8 km from the release point for normal operations and Category 1 event sequences is determined by summing up the mean value of the parameter "L-BONE SUR TOT LIF" and the ingestion dose. The TODE at 11 km or 8 km from the release point for Category 2 event sequences is determined by summing up the 95th percentile value of the parameter "L-BONE SUR TOT LIF" and the ingestion dose.

The onsite public TEDE at 100 m from the release point for normal operations and Category 1 event sequences do not include ingestion dose and are calculated based on an exposure period of 2,000 hr (Section 4, Assumption 4.20). Therefore, the annual TEDE needs to be adjusted to reflect an exposure period of 2,000 hr instead of 8,760 hr.

The onsite public TEDE, at 100 m from the release point for normal operations, is the mean value of the parameter "L-EFFECTIVE TOT LIF" in cohorts 1 and 2 (i.e., combined EARLY and CHRONC outputs) multiplied by a factor of 0.228, which is equal to 2,000 hr divided by 8,760 hr.

The onsite public TEDE, at 100 m from the release point for Category 1 event sequences, is determined by summing up the mean value of the parameter "L-EFFECTIVE TOT LIF" in cohort 1 (the EARLY phase) and the parameter "L-EFFECTIVE TOT LIF" in cohort 2 (the CHRONC phase) multiplied by a factor of 0.228.

6.2.1.2 Source Terms

Source term input parameters are needed in the ATMOS file. Source term is defined as concentrations or inventories of radionuclides in SNFs or waste forms to be received and handled at the repository. The concentration or inventory of each radionuclide in a waste form is expressed as curies per SFA, curies per unit weight of waste form, or curies per canister. A curie is a unit quantity of any radioactive nuclide in which 3.7×10^{10} disintegrations occur per second. Source terms for waste forms to be received and handled at the repository are needed for consequence analyses. Source terms are a function of the initial fuel enrichment, fuel compound, cladding type, moderator type, and reactor operating history. Source terms are calculated using a computer program that performs a point depletion and decay calculation.

Commercial Spent Nuclear Fuel–Average and maximum source terms for PWR and BWR commercial SNF are from BSC (2004l) and BSC (2003a).

The SAS2H sequence in SCALE V4.3, from NUREG/CR-0200 (ORNL 1997), is used to calculate the PWR and BWR source terms for selected SFAs as a function of assembly average burnup and cooling time. The prime functional module of the SAS2H code sequence used is the ORIGEN-S module. This module performs a point depletion and decay calculation of a selected fuel type with user-specified irradiation conditions and decay times. The resulting source terms are then extracted from the SAS2H output and are used as input to consequence analyses. Source terms for PWR and BWR SFAs with four different combinations of initial enrichment, burnup, and decay time are considered in this consequence analysis and are presented in Table 3.

For Category 1 event sequences, both Average PWR and Average BWR SFAs are used to calculate mean doses and 50^{th} percentile doses. The calculated mean dose is used because it is higher than the calculated 50^{th} percentile dose. For Category 2 event sequences, both Maximum PWR SFAs and Maximum BWR SFAs are used to calculate doses under 95^{th} percentile weather conditions in accordance with Regulatory Guide 1.145, p. 3. Radionuclide inventories in Ci/FA are presented in Table 4 for each nuclide and fuel type evaluated.

Crud (Commercial SNF)–Crud is activated corrosion products found on the exterior surface of SFAs, primarily caused by the irradiation of reactor internals and imperfect water chemistry control in a reactor coolant system. Crud can be released to the environment during an accident involving commercial SNF. After decaying for 5 years, the nuclide species that have significant activity in the crud are ^{55}Fe and ^{60}Co. The commercial SNF assemblies have initial crud activities at the time of discharge from the reactor as shown in Table 12.

Table 12. Commercial Spent Nuclear Fuel Assembly Initial Crud Activities

Radionuclide	PWR (μCi/cm^2)	BWR (μCi/cm^2)
^{60}Co	140	1254
^{55}Fe	5902	7415

NOTE: Crud activities are bounding estimates based on analysis of measured crud activity data (BSC 2004m, Table 4).

BWR = boiling water reactor; PWR = pressurized water reactor; SNF = spent nuclear fuel.

The crud surface activity for a given assembly is a function of time after discharge from the reactor. The time-dependent crud surface activity is based on the following radioactive decay equation (BSC 2004l, p. 27):

$$N(t) = N(0) \exp(-t \times \ln 2 / t_{1/2}) \qquad \text{(Eq. 14)}$$

where,

$N(t)$ = crud activity at time t,
$N(0)$ = crud activity at time 0,
$t_{1/2}$ = radionuclide half-life in years
= 5.271 years for ^{60}Co (Eckerman and Ryman 1993, Table A.1), and 2.73 years for ^{55}Fe (BSC 2004l, Table 6)
t = the decay time in years.

The crud source term (Ci/FA) released to the environment, on a per assembly basis, is calculated as:

$$ST_{crud} = SA_{crud} \times A_{SFA} \times conv \qquad \text{(Eq. 15)}$$

where,

ST_{crud} = crud source term; Ci/FA
SA_{crud} = crud surface activity; μCi/cm^2
A_{SFA} = surface area per assembly; cm^2/FA
$conv$ = conversion factor; 10^{-6} Ci/μCi.

Commercial SNF fuel assemblies have the following surface areas, A_{SFA}:

- PWR = 449,003 cm^2/assembly (BSC 2004l, p. 27)
- BWR = 168,148 cm^2/assembly (BSC 2003a, Table 45).

These surface areas are bounding estimates based on SFAs with the highest known surface areas, which are a South Texas PWR assembly (BSC 2004l, p. 27) and an ANF 9 × 9 JP-4 BWR assembly (BSC 2003a, Table 45). The crud source term for Category 2 event sequences is based on Maximum PWR or Maximum BWR SNF with a 5-year decay time. Using Equations 14 and 15, the 5-year crud source term is given in Table 13. The crud source term for Category 1 event sequences is based on Average PWR or Average BWR SNF with a 25-year decay time. Using Equations 14 and 15, the 25-year crud source term is given in Table 13.

Table 13. Five-year and 25-year Crud Source Terms

Radionuclide In Crud	5-year Crud Source (Ci/FA)	25-year Crud Source (Ci/FA)
Fe-55 PWR	7.45E+02	4.64
Fe-55 BWR	3.50E+02	2.18
Co-60 PWR	3.26E+01	2.35
Co-60 BWR	1.09E+02	7.87

BWR = boiling water reactor; Ci/FA = curies per fuel assembly;
PWR = pressurized water reactor.

DOE Spent Nuclear Fuel–DOE SNF is shipped in two types of disposable canisters, namely, the DOE standardized canister and the multicanister overpack. DOE SNF is received at the repository in transportation casks containing the sealed disposable canisters. ISG-5 (NRC 2003a, Attachment, p. 3) states that for casks having closure lids that are designed and tested to be leak tight as defined in ANSI N14.5-97, detailed consequence analyses are not necessary.

Because the DOE standardized canister and multicanister overpack are designed and tested to the ANSI N14.5-97 leak-tight standard after dropping from a design basis drop height, a drop and breach of DOE canisters or multicanister overpacks is a beyond Category 2 event (BSC 2004k, Table 16) and detailed consequence analyses are not necessary as stated in ISG-5 (NRC 2003a, Attachment, p. 3).

Naval Spent Nuclear Fuel–Some drops or collisions involving a naval canister may result in a breach of the canister but no fuel damage. These events do not correspond to any bounding Category 2 event sequences listed in Table 10, because they do not cause fuel damage. For this category of events, a release source term based on radionuclides involving only crud was developed (Gisch 2004, p. 1). A table of radionuclides, for crud and total activity, for a representative naval SNF canister 5 years after shutdown is provided in Gisch (2004, Table 1). The crud inventory for naval SNF is calculated using the standard naval program shielding procedure. This value was increased by a factor of 2.5 to provide a conservative crud concentration for use in developing the source term. Table 14 identifies a radionuclide release source term for Category 2 event sequences in which a naval SNF canister is breached and fuel is not damaged.

A conservative damage ratio, along with the crud ARF and RF (Table 5) were applied to the naval crud inventory, using a conservative leak path factor of 1.0 (Section 4, Assumption 4.11). The release source term in Table 14 takes no credit for HEPA filtration, plateout, or deposition of radionuclides.

Table 14. Crud Release From a Naval Spent Nuclear Fuel Canister

Radionuclide	Activity (Ci)
Am-241	2.7E-06
Am-242m	1.5E-08
Am-243	2.3E-08
C-14	7.8E-03
Cm-242	2.1E-08
Cm-243	1.7E-08
Cm-244	2.3E-06
Cm-245	2.0E-10
Cm-246	7.8E-11
Cm-247	2.3E-16
Cm-248	7.4E-16
Co-60	4.1E-01
Cs-137	2.8E-04
Fe-55	4.4E-01
I-129	3.1E-08
Nb-93m	9.5E-03
Nb-94	1.6E-04
Ni-59	2.3E-03
Ni-63	2.3E-01
Np-237	2.3E-11
Pu-238	1.9E-06
Pu-239	3.1E-07
Pu-240	2.0E-07
Pu-241	6.1E-05
Pu-242	2.3E-09
Se-79	1.2E-09
Sn-126	3.5E-09
Sr-90	2.8E-04
Tc-99	7.8E-06
Th-232	7.4E-13
U-232	1.1E-08
Zr-93	1.6E-06

Source: Gisch 2004, Table 1

Table 15 identifies a source term, based on radionuclide inventories involving releases from crud and fuel-bearing regions for bounding Category 2 event sequences (Table 10) that could result in a canister breach with fuel damage. These Category 2 event sequences involve slapdowns onto unyielding surfaces resulting in the highest canister strain levels and the highest potential for fuel damage, because of contact between the SFAs and the canister basket support plates.

Table 15. Fuel and Crud Release From a Naval Spent Nuclear Fuel Canister

Nuclide	Activity – All Pathways Except Inhalation (Ci)	Activity – Inhalation Pathway Only (Ci)
Ac-227	2.9E-11	1.5E-13
Am-241	1.8E-05	2.8E-06
Am-242m	1.5E-07	1.6E-08
Am-243	2.2E-07	2.4E-08
Ba-137m	1.3E-01	6.4E-04
C-14	7.8E-03	7.8E-03
Cd-113m	1.1E-05	5.6E-08
Cf-252	3.5E-13	1.7E-15
Cm-242	4.5E-07	2.4E-08
Cm-243	2.5E-07	1.9E-08
Cm-244	2.1E-05	2.4E-06
Cm-245	2.3E-09	2.1E-10
Cm-246	4.8E-10	8.0E-11
Cm-247	3.0E-15	2.5E-16
Cm-248	8.5E-15	7.8E-16
Co-60	4.1E-01	4.1E-01
Cs-134	1.7E-01	1.7E-01
Cs-135	9.3E-06	9.3E-06
Cs-137	9.0E-01	9.0E-01
Eu-154	3.4E-03	1.7E-05
Eu-155	5.6E-04	2.8E-06
Fe-55	4.4E-01	4.4E-01
H-3	4.2E+00	4.2E+00
I-129	3.7E-04	3.7E-04
Kr-85	1.1E+02	1.1E+02
Nb-93m	9.5E-03	9.5E-03
Nb-94	2.1E-04	1.6E-04
Ni-59	2.3E-03	2.3E-03
Ni-63	2.3E-01	2.3E-01
Np-237	4.7E-07	2.4E-09
Pa-231	1.5E-10	7.7E-13
Pb-210	2.7E-13	1.3E-15
Pd-107	1.8E-08	9.0E-11
Pm-147	4.8E-02	2.4E-04
Pu-238	3.5E-03	1.9E-05
Pu-239	3.9E-06	3.3E-07
Pu-240	4.2E-06	2.2E-07
Pu-241	1.3E-03	6.7E-05
Pu-242	2.6E-08	2.5E-09
Ra-226	1.6E-12	8.2E-15
Ra-228	1.5E-16	7.5E-19
Rh-102	8.3E-09	4.2E-11
Ru-106	1.2E-02	1.2E-02

Table 15. Fuel and Crud Release From a Naval Spent Nuclear Fuel Canister (Continued)

Nuclide	Activity – All Pathways Except Inhalation (Ci)	Activity – Inhalation Pathway Only (Ci)
Sb-125	9.8E-04	4.9E-06
Se-79	1.0E-07	1.7E-09
Sm-147	9.6E-12	4.8E-14
Sm-151	4.3E-04	2.1E-06
Sn-126	3.6E-07	5.3E-09
Sr-90	1.3E-01	9.4E-04
Tc-99	2.9E-05	7.9E-06
Th-229	2.7E-12	1.3E-14
Th-230	5.5E-10	2.7E-12
Th-232	7.4E-13	7.4E-13
U-232	1.8E-07	1.2E-08
U-233	7.1E-10	3.6E-12
U-234	4.5E-06	2.2E-08
U-235	8.6E-08	4.3E-10
U-236	7.5E-07	3.7E-09
U-238	3.6E-10	1.8E-12
Y-90	1.3E-01	6.7E-04
Zr-93	3.8E-06	1.6E-06

Source: Gisch 2004, Table 2

The fuel and crud damage ratios, used to develop the source term, were conservatively selected to ensure that canister radionuclide inventories and internal hardware designs are bounded. ARFs and RFs for standard commercial SNF (Table 5) are used for Category 2 event sequences involving naval SNF. A conservative leak path factor of 1.0 was used (Section 4, Assumption 4.11). Separate activity columns are listed: one for dose calculations for all pathways except inhalation, and one that applies the RFs in Table 5 for dose calculations in the inhalation pathway. The release source term in Table 15 takes no credit for HEPA filtration, plateout, or deposition of radionuclides.

Vitrified HLW–Vitrified HLW forms from the Savannah River Site (SRS), Hanford Site, West Valley, and INEEL are received at the repository in sealed canisters inside transportation casks. The per canister source terms for vitrified HLW from the four sites are provided in Table 16.

The per canister source term for vitrified HLW shipped from SRS (Table 16) is from Fowler (2003, Table 2, column 3); this source term is the projected maximum radionuclide inventory per canister based on waste tank sludge batches collected as of March 17, 2003.

Table 16. Radionuclide Inventory Per Vitrified High-level Radioactive Waste Canister

Nuclide	SRS (Ci)[a]	Hanford (Ci)[b]	West Valley (Ci)[c]	INEEL (Ci)[d]
Am-241	2.28E+02	4.65E+02	5.07E+02	2.61E+00
Am-242m	1.30E-01	0.00E+00	2.74E+00	0.00E+00
Am-243	3.68E-01	9.99E-02	3.28E+00	0.00E+00
Ba-137m	6.24E+04	6.62E+04[e]	3.00E+04[e]	1.40E+04[e]
C-14	0.00E+00	1.06E-07	1.30E+00	0.00E+00
Cd-113m	0.00E+00	2.69E+01	0.00E+00	0.00E+00
Ce-144	1.80E+00	0.00E+00	0.00E+00	1.23E+02
Cm-242	0.00E+00	3.46E-01	0.00E+00	0.00E+00
Cm-243	4.14E-01	4.42E-02	0.00E+00	0.00E+00
Cm-244	1.86E+03	4.27E-01	5.75E+01	5.48E-01
Cm-245	1.49E-01	0.00E+00	0.00E+00	0.00E+00
Cm-246	4.34E-02	0.00E+00	0.00E+00	0.00E+00
Co-60	7.15E+02	1.04E+00	0.00E+00	0.00E+00
Cs-134	2.40E+02	2.23E+02	0.00E+00	7.85E+02
Cs-135	2.62E-01	0.00E+00	1.09E+00	0.00E+00
Cs-137	6.67E+04	7.00E+04	3.17E+04	1.48E+04
Eu-152	0.00E+00	4.96E+00	0.00E+00	0.00E+00
Eu-154	1.68E+03	7.92E+00	0.00E+00	1.54E+02
Eu-155	6.70E-01	3.28E+02	0.00E+00	0.00E+00
I-129	7.35E-05	0.00E+00	0.00E+00	0.00E+00
Nb-93m	1.47E-01	2.44E+00	1.96E+00	0.00E+00
Ni-59	2.16E-01	4.96E-01	1.00E+00	0.00E+00
Ni-63	1.41E+01	5.13E+01	7.74E+01	0.00E+00
Np-237	3.39E-02	2.50E-01	1.50E-01	0.00E+00
Pa-231	0.00E+00	4.24E-04	1.44E-01	0.00E+00
Pm-147	4.86E+03	0.00E+00	0.00E+00	4.09E+03
Pu-238	5.93E+03	2.29E+00	3.92E+01	8.60E+01
Pu-239	4.90E+01	2.13E+01	8.75E+00	8.92E-01
Pu-240	3.34E+01	6.42E+00	6.28E+00	8.27E-01
Pu-241	3.49E+03	1.22E+02	3.11E+02	1.61E+02
Pu-242	1.11E-01	9.91E-04	8.15E-03	0.00E+00
Ru-106	4.70E+00	1.61E-02	0.00E+00	4.07E+01
Sb-125	0.00E+00	1.87E+01	0.00E+00	0.00E+00
Se-79	5.91E-02	9.15E-02	5.70E-01	0.00E+00
Sm-151	1.22E+02	3.62E+03	7.63E+02	0.00E+00
Sn-126	3.09E-02	5.74E-01	9.85E-01	0.00E+00
Sr-90	7.62E+04	7.38E+04	2.80E+04	1.52E+04
Tc-99	1.56E+01	2.31E+01	8.72E+00	0.00E+00
U-232	3.68E-04	4.72E-04	0.00E+00	0.00E+00
U-233	2.75E-02	2.09E-03	9.03E-02	0.00E+00
U-234	8.10E-02	1.46E-02	2.40E-02	0.00E+00
U-235	6.00E-04	5.56E-04	0.00E+00	0.00E+00
U-236	7.54E-03	1.18E-03	0.00E+00	0.00E+00

Table 16. Radionuclide Inventory Per Vitrified High-level Radioactive Waste Canister (Continued)

Nuclide	SRS (Ci)[a]	Hanford (Ci)[b]	West Valley (Ci)[c]	INEEL (Ci)[d]
U-238	5.17E-02	1.01E-02	0.00E+00	0.00E+00
Y-90	7.62E+04	7.38E+04	2.80E+04	1.52E+04
Zr-93	1.88E-01	5.76E+00	2.58E+00	0.00E+00

NOTES: [a] Fowler 2003, Table 2, column 3
[b] DOE 2004c, Table 5, column 4
[c] WVNS 2001, Table 5, column 4 (high estimate case for 1996)
[d] CRWMS M&O 1999b, p. III-1
[e] Ba-137m = 0.946 × Cs-137 (Eckerman and Ryman 1993, p. 203).

INEEL = Idaho National Engineering and Environmental Laboratory; SRS = Savannah River Site.

The per canister source term for vitrified HLW shipped from the Hanford Site (Table 16) is from *Waste Treatment and Immobilization Plant (WTP) High-Level Waste (HLW) Canister Production Estimates to Support Analyses by the Yucca Mountain Project* (DOE 2004c, Table 5, column 4); this source term is the estimated bounding radionuclide inventory per canister based on HLW canisters produced from the 241-AZ-101 waste tank decayed to January 1, 2010. Among the waste tanks in the Hanford tank farm, the 241-AZ-101 waste tank has been determined to be the waste tank that will contain the maximum concentration of radionuclides for the Hanford HLW canisters (DOE 2004c, p. 6).

The per canister source term for vitrified HLW (Table 16), shipped from West Valley, is the estimated highest radionuclide inventory per canister, based on a 100 percent fill canister with radionuclides decayed to the year 1996, from *West Valley Nuclear Services Company Waste Form Qualification Report* (WVNS 2001, Table 5, column 4).

The per canister source term for vitrified HLW shipped from INEEL (Table 16) is from *DOE High-Level Vitrified Waste Dose Calculation* (CRWMS M&O 1999b, p. III-1). Dose calculations show that SRS HLW has the highest dose consequence of HLW from the four DOE sites, those being Hanford, INEEL, SRS, and West Valley (CRWMS M&O 1999b, p. III-1).

6.2.1.3 Release Fractions

Radionuclide release fractions are needed in the ATMOS file. The Category 1 and Category 2 event sequence total release fraction is defined as the fraction of total inventory of a given radionuclide that is released to the environment from a commercial SNF element following an event sequence; for example, drop of a fuel element. The release fraction for commercial SNF is primarily a measure of the inventory of fuel particulates, gases, and volatile species present in a breached fuel element.

The source term released from Category 1 and Category 2 event sequences are a function of the MAR, DR, ARF, RF, $(LPF)_{cask}$, $(LPF)_{fac}$, and $(LPF)_{HEPA}$:

Release Source Term $ST_j = MAR_j \times DR \times ARF \times RF \times (LPF)_{cask} \times (LPF)_{fac} \times (LPF)_{HEPA}$ (Eq. 16)

The MAR_j is the material at risk for radionuclide j. The DR is the damage ratio or the fraction of fuel rods that are assumed to fail by cladding breach during an event sequence. The ARF is the fraction of the total radionuclide inventory in damaged fuel rods that is released from breached cladding and is suspended in air as an aerosol following an event sequence. The RF can be transported through air, inhaled into the human respiratory system, and contribute to the inhalation dose.

The $(LPF)_{cask}$ is the fraction of the ARF that reaches the ventilation system after local deposition from plate-out and gravitational settling inside a cask. The $(LPF)_{fac}$ is the fraction of the ARF that reaches the ventilation system after local deposition from plate-out and gravitational settling inside a facility. The $(LPF)_{HEPA}$ is the fraction of radionuclides that is released to the environment after discharging from the HEPA filters in a surface facility ventilation system.

For Category 1 and Category 2 event sequences, the DR is assumed to be 1.0 for commercial SNF and for HLW in a canister (Section 4, Assumption 4.10). For Category 2 event sequences, the $(LPF)_{cask}$ is assumed to be 0.1 for SNF in a transportation cask and 0.01 for HLW in a canister in a transportation cask (Section 4, Assumption 4.11). For normal operations and Category 1 event sequences, the $(LPF)_{fac}$ is assumed to be 1.0 for a waste transfer cell (Section 4, Assumption 4.12).

Commercial Spent Nuclear Fuel Release Fractions–For commercial SNF releases in air, ARF and RF parameters for Category 1 and Category 2 dose assessments are shown in Table 5.

An intact commercial SNF assembly has intact cladding and structural integrity before a Category 1 or Category 2 event occurs. There are four types of canistered commercial SNF with a loss of its original assembly structure, with damaged cladding, or with both, as follows (BSC 2003c, pp. G-1 and G-2):

- Type 1: Mechanically and cladding-penetration damaged commercial SNF
- Type 2: Consolidated or reconstituted SFAs, or both
- Type 3: Fuel rods, pieces, and debris
- Type 4: Non-fuel components.

Type 3 is further divided into (3a) fuel rods with intact cladding and (3b) other fuel rods, pieces, and debris.

Type 1 and Type 3b contain commercial SNF that are classified as failed commercial SNF by waste generators. Known or suspected failed fuel is segregated from intact fuel at nuclear power plants. Failed SFAs and fuel debris are placed into either unsealed screen-end canisters or sealed solid-end canisters (BSC 2003c, p. G-2). Drops or collisions of canistered commercial SNF rods in transportation casks and waste packages are assumed to be Category 2 event sequences (Section 4, Assumption 4.9). In Table 5, only the ARF and RF values for Type 3a (fuel rods with intact cladding) are presented because these values bound the ARF and RF values for fuel types 1, 2, 3b, and 4.

The ARFs and RFs for intact commercial SNF assemblies (Table 5) are used for normal operations and Category 1 event sequences. The ARFs and RFs for Type 3a (fuel rods with intact cladding) are used for Category 2 event sequences because the combined ARF $\times$ RF values for Type 3a are larger or equal to the values for intact commercial SNF assemblies. Because drops or collisions of Type 3a in transportation casks and waste packages are assumed to be Category 2 event sequences (Section 4, Assumption 4.9), the ARFs and RFs for Type 3a are not applicable to normal operations and Category 1 event sequences.

Damaged commercial SNF in damaged fuel cans will be shipped to the repository inside a transportation cask as stated in ISG-1 (NRC 2002). ISG-1 allows utilities to ship fuel with pinhole leaks or hairline cracks as intact fuel (NRC 2002). Intact fuel, damaged fuel in canisters, and fuel with pinhole leaks or hairline cracks will be transferred, from a transportation cask to a waste package, inside a waste transfer cell during normal operations in a DTF or the FHF. Exposing the failed fuel to air, while inside a DTF or the FHF, could cause oxidation of the fuel (UO_2) to higher oxides, such as U_3O_8. When U_3O_8 starts to form, the fuel pellet volume expands and eventually cladding could unzip.

The oxidation of SNF in air, as a two-step process of the form $UO_2 \rightarrow UO_{2.4} \rightarrow U_3O_8$, is described in Hanson (1998, p. iii). The transition from $UO_2 \rightarrow UO_{2.4}$ does not result in appreciable density changes. The transition from $UO_{2.4} \rightarrow U_3O_8$ results in a 20 percent less dense phase. The increase in volume as SNF oxidizes to U_3O_8 places stress on the cladding, which may split as a result.

The oxidation process first progresses by the $UO_2 \rightarrow UO_{2.4}$ reaction. Once the SNF oxidized to $UO_{2.4}$, a plateau is reached where the fuel resisted oxidation to higher oxides. Following this plateau oxidation resumes until the U_3O_8 phase is reached. Hanson (1998, p. iii) found that the $UO_{2.4} \rightarrow U_3O_8$ reaction is strongly dependent on both temperature and burnup. SNF strongly resisted oxidation beyond $UO_{2.4}$ at either low temperature or high burnup.

Little or no oxidation is expected to occur for intact fuel or fuel with pinhole or hairline cracks during fuel handling operations in the repository (Section 4, Assumption 4.25). The fuel fine release fraction of 3×10^{-5} for a burst rupture is conservatively used for fuel with pinhole leaks or hairline cracks where little or no fuel oxidation occurs (Section 4, Assumption 4.25).

In addition to 1 percent of the fuel rods received at the repository having pinhole leaks or hairline cracks (Section 4, Assumption 4.3), it is assumed that 154 failed fuel rods (Section 4, Assumption 4.23) received at the repository per year are expected to be vulnerable to oxidation and cladding unzipping. An ARF of 1.2×10^{-4} is assumed for fuel rods expected to be vulnerable to oxidation and cladding unzipping (Section 4, Assumption 4.24).

In summary, for fuel rods with pinhole or hairline crack failures, a fuel fine release fraction of 3×10^{-5} (Section 4, Assumption 4.25) (Table 5) is used to calculate the normal operation doses. For the purposes of calculating doses it is assumed that 1 percent of the total fuel rods received at the repository have pinhole or hairline crack failures.

In summary, for fuel rods with cladding failures greater than pinhole or hairline cracks, a fuel fine release fraction of 1.2×10^{-4} (Section 4, Assumption 4.24) (Table 5) is used to calculate the normal operation doses. This release fraction is applicable to 154 fuel rods per year (Section 4, Assumption 4.23) that have cladding failures greater than pinhole or hairline cracks.

It is assumed that the fission product gas, volatile species, and crud release fractions for breaches of intact commercial SNF assemblies and rods in Table 5 are applicable to releases of fission product gases, volatile species, and crud during oxidation of damaged commercial SNF in air in Table 5 (Section 4, Assumption 4.16).

HLW Release Fractions–The formation of particulates from an impact breach of a HLW canister is based on ANSI/ANS-5.10-1998, p. 15. The results are based on empirical measurements of impact tests on UO_2, ceramic, and glass-simulated waste forms. Small-scale laboratory tests establish each correlation for the percentage of respirable size release fractions created during impacts.

This method uses the available results to develop a method to estimate the ARFs of HLW canister glass that could be released as airborne particulates. Based on these methods, the release fraction of respirable airborne particulates or the release fraction pulverized into respirable sizes (< 10 μm) from a drop event (PULF) formed following an impact is estimated as:

$$\text{PULF} = 2\text{E-4 cm}^3/\text{joule} * \text{E/V} \qquad \text{(Eq. 17)}$$

where,

PULF	=	release fraction pulverized into respirable sizes (< 10 μm) from a drop event (units = dimensionless)
E/V	=	impact energy density in impacted HLW
	=	1.0E-07 joule-s^2/g-cm^2 * ρ * g * h (CRWMS M&O 1999b, Attachment V)

where,

ρ	=	density of the HLW dropped
	=	2.75 g/cm^3 (CRWMS M&O 1999b, Section 5.2.5)
g	=	gravitational constant
	=	980.7 cm/s^2 (CRWMS M&O 1999b; Attachment V)
h	=	drop height in cm.

<u>PULF for HLW Canister Drop</u>

From Equation 17:

PULF = 2E-4 cm^3/joule * 1.0E-7 joule-s^2/g-cm^2 * 2.75 g/cm^3 * 980.7 cm/s^2 * (drop height) cm

Calculated PULF for selected drop heights (CRWMS M&O 1999b; Attachment V) are:

Drop Height	PULF
80 in. (203 cm)	1.10E-05
264 in. (671 cm)	3.62E-05
330 in. (838 cm)	4.52E-05
448 in. (1138 cm)	6.14E-05

The drop height of 448 in. represents the worst case with a PULF of 6.14E-05, which is used in the public dose calculations for HLW canisters because it bounds the other values. It is assumed that the canister handling system is designed so that a drop of an HLW canister, inside a surface facility, will not exceed a drop height of 276 in. (23 ft) (Section 4, Assumption 4.19) and, therefore, the use of a 448-in. drop height is conservative.

Naval Spent Nuclear Fuel Release Fractions–ARFs and RFs used for Category 2 event sequences involving naval SNF are presented in Section 6.2.1.2.

HEPA Filter Leak Path Factor–The HEPA filter $(LPF)_{HEPA}$ refers to the removal of particulates provided by HEPA filters present in the surface facility ventilation system. For normal operations and Category 1 event sequences, a two-stage HEPA filtration system with a particulate removal efficiency of 99 percent per stage (i.e., a HEPA LPF of 0.01/stage) is assumed (Section 4, Assumption 4.8), which is consistent with NRC-recommended credit for accident dose evaluations in Regulatory Guide 1.140, p. 1.140-4, and Regulatory Guide 1.52, p. 1.52-5, for a combined efficiency of 99.99 percent for two stages; a HEPA LPF of 10^{-4}.

It is further assumed that the HVAC system removes particulates and cesium in air through two stages of HEPA filters in series, which are protected by prefilters, sprinklers, and demisters (Section 4, Assumption 4.8).

An $(LPF)_{HEPA}$ of 10^{-4} is more conservative than the value of 2×10^{-6} for two stages of HEPA filters, as discussed in NUREG/CR-6410 (SAIC 1998, Section F.2.1.3) (Section 4, Assumption 4.8). NUREG/CR-0722 test data (Lorenz et al. 1980, Table 19) show that the two-stage HEPA filters capture almost 100 percent of incoming airborne cesium.

No credit is taken for HEPA filters in mitigating radiological consequences from Category 2 event sequences. Credit is taken for HEPA filters in mitigating radiological consequences from normal operations and Category 1 event sequences (Section 4, Assumption 4.4).

For normal operations, and Category 1 and Category 2 event sequences, no credit is taken for charcoal adsorbers to remove radionuclides (Section 4, Assumption 4.18).

6.2.1.4 Dose Conversion Factors

In MACCS2 (ORNL 1998), DCFs for inhalation, cloudshine, groundshine, and ingestion pathways are generated by the FGRDCF module.

The output filename generated by the FGRDCF module is YMPD825.INP. This file is called by the EARLY file and the CHRONC file during MACCS2 execution. The YMPD825.INP file contains DCF data from Federal Guidance Reports Nos. 11 (Eckerman et al. 1988) and 12 (Eckerman and Ryman 1993) for inhalation, cloudshine, groundshine, and ingestion pathways, in accordance with NUREG-1804 (NRC 2003b, Section 2.1.1.5).

DCFs for inhalation are dependent on the chemical form of the radionuclide, which is represented by the lung clearance class (D = daily, W = weekly, Y = yearly) and the fractional uptake from the small intestine to blood (f1). Some nuclides have only one lung clearance class, for example, ^{3}H, whereas others have multiple lung clearance classes, for example, ^{239}Pu.

The inhalation DCFs used by YMPD825.INP are from Federal Guidance Report No. 11 (Eckerman et al. 1988, Table 2.1). The air submersion DCFs for gonads, breast, lungs, red marrow, bone surface, thyroid, remainder, effective–whole body, and skin used by YMPD825.INP are from Federal Guidance Report No. 12 (Eckerman and Ryman 1993, Table III.1). The remainder is a weighted combination of five remaining organs or tissues receiving the highest doses, such as the liver, kidneys, spleen, brain, small intestine, upper large intestine, lower large intestine, or other organs, excluding the skin, lens of the eye, and extremities (Eckerman et al. 1988, p. 6).

6.2.1.5 Location of Maximally Exposed Individual

The maximally exposed individual is defined as an individual located at a distance that corresponds to the approximate distance between the surface facility or the subsurface repository and the nearest point of public access on the repository site boundary, which lies to the west (Section 4, Assumption 4.15).

This individual is assumed to be present at the site boundary and exposed to ground contamination and intake of contaminated food for one year (Section 4, Assumptions 4.5 and 4.6).

A site boundary distance of 11 km (Attachment B, Figure B-1) is used to calculate χ/Q values from radiological releases from the surface facility. This distance corresponds to the distance from the DTF ventilation exhaust shaft to the nearest point on the site boundary that is the closest point where any member of the public could be standing, or living, at the time of a postulated radiological release (Section 4, Assumption 4.15).

A site boundary distance of 8 km (Attachment B, Figure B-1) is used to calculate χ/Q values from radiological releases from the subsurface repository exhaust shafts. This distance corresponds to the approximate distance between the subsurface repository and the nearest point of public access on the site boundary, which lies to the west (Section 4, Assumption 4.15).

Attachment B (Figures B-1 and B-2) shows the locations of the site boundaries, restricted areas, and preclosure dose limits. Doses to members of public onsite were calculated using MACCS2 at various distances from the release points. For releases from the surface or subsurface facilities, the maximum dose to members of the onsite public occurs at the shortest distance from the release point. Because members of the public can be at any distance outside the restricted area and within the site boundary, the dose to members of the public is calculated at 100 m.

The maximum dose for subsurface releases occurs at the shortest distance from the subsurface facility exhaust shafts. Doses to members of the public at 100 m and 3 km for releases from a subsurface exhaust shaft are reported in Section 6.3.1.

The dose to members of the public at 3 km from subsurface facility releases is used in dose aggregation with the dose to members of the public at 100 m from surface facility releases in Section 6.3.2.

The maximally exposed individual at the site boundary of 11 km is assumed to receive doses from the inhalation, resuspension inhalation, air submersion, groundshine, and ingestion pathways for a period of 8,760 hr (Section 4, Assumption 4.20).

The onsite individual member of the public, at 100 m or 3 km away from a DTF, the FHF, or a subsurface exhaust shaft, is assumed to receive doses from inhalation, resuspension inhalation, air submersion, and groundshine pathways for a period of 2,000 hr (Section 4, Assumption 4.20).

6.2.1.6 Food Consumption and Production Rates

The consequence analysis using MACCS2 (ORNL 1998) requires that any farmland used in the 50-mi radius from the point of a radioactive release be defined. Annual consumption rates of foods are from a 1997 regional survey of the population in the vicinity of Amargosa Valley, as described in *Characteristics of the Receptor for the Biosphere Model* (BSC 2004r).

This analysis is conducted using the 1997 survey of food consumption rates for each food type, as shown in Table 17. These values are entered into input files for the COMIDA2 module.

The COMIDA2 module uses the food consumption rates in Table 17 to generate the food pathway factor data file YUCCALA.BIN. Because the food production rates are not used in this calculation to calculate dose to maximally exposed individual members of the public from foods, the values of these input parameters are not discussed.

This file is believed to provide conservative estimates of the individual dose from food and the total population dose from foods. The food pathway file name is specified in the CHRONC input file by keyword BIN_FILE.

Table 17. Food Consumption Rates Used as Input to COMIDA2

Food Product	Intake (kg/year)	COMIDA2 Category
Cereal and Bread	0.23	Grains
Eggs; eggs and fish	5.53	Other
Fruit	12.68	Fruits
Leafy Vegetables	3.78	Leafy vegetables
Meat; beef, pork and wild game	2.85	Beef
Milk	4.66	Milk
Other Vegetables	4.73	50 percent split to Roots and Legumes
Poultry	0.42	Poultry

Source: BSC 2004r, Table 6-21

6.2.2 Worker Dose Calculation Design Inputs

This section discusses input parameters used in worker dose calculations for normal operations and Category 1 event sequences.

The source terms, release fractions, and DCFs used as input to worker dose calculations are the same as those used in public dose calculations (Sections 6.2.1.2 to 6.2.1.4).

Other input parameters, such as surface and subsurface facility dimensions, and locations, such as receptors, exhaust stacks, and exhaust shafts, are discussed in airborne release and direct radiation worker dose calculations (BSC 2004a through 2004j).

6.2.3 Regulations

Regulation 10 CFR 63.111(a) states that the repository operations area must meet the requirements of 10 CFR Part 20, and that the annual TEDE resulting from normal operations and Category 1 event sequences to any real member of the public located beyond the site boundary must meet the preclosure standard specified in 10 CFR 63.204.

In digest, 10 CFR 63.111(b) states:

- The sum of doses from normal operations and Category 1 event sequences must meet the requirements of 10 CFR 63.111(a)
- For single Category 2 event sequences, the public dose at or beyond the site boundary may not exceed the more limiting of a TEDE of 5 rem, or the sum of the DDE and the CDE to any individual organ or tissue, other than the lens of the eye, of 50 rem
- For single Category 2 event sequences, the LDE may not exceed 15 rem and the SDE may not exceed 50 rem at or beyond the site boundary.

Regulation 10 CFR 20.1201 states that the licensee shall control the occupational dose to individual adults to the following dose limits:

- An annual dose limit, which is the more limiting of the TEDE of 5 rem, or the sum of the DDE and the CDE of 50 rem to any individual organ or tissue other than the lens of the eye
- Annual limits, which are a LDE of 15 rem and a shallow dose equivalent of 50 rem to skin or to any extremity.

Regulation 10 CFR 20.1301 states that each licensee shall conduct operations so that:

- The TEDE to individual members of the public from the licensed operation does not exceed 0.1 rem in a year
- The dose in any unrestricted area from external sources does not exceed 2 mrem in any 1 hr.

Radiation dose limits for Category 1 and Category 2 event sequences are specified in 10 CFR 63.111(b)(1) and 10 CFR 63.111(b)(2). Regulation 10 CFR 63.111(b)(1) references 10 CFR 63.111(a)(1), which references 10 CFR Part 20. Regulation 10 CFR 63.111(a)(2) references 10 CFR 63.204. These regulations specify: (1) dose limits for workers and members of the public during normal operations and for Category 1 event sequences, which are summarized in Table 1, and (2) public dose limits for Category 2 event sequences, which are summarized in Table 2.

In Tables 1 and 2, the dose receptors are divided into three groups, namely, workers, onsite members of the public, and offsite members of the public. For dose compliance purposes, the people who work or visit at the Yucca Mountain site or work near the site are assigned to one of these three groups. The preclosure dose limits for workers in the restricted area; onsite members of the public in the preclosure controlled area, and offsite members of the public in the unrestricted area are shown in Attachment B (Figures B-1 and B-2).

6.3 PUBLIC DOSE RESULTS

This section discusses the results of MACCS2 (ORNL 1998) calculations of doses to members of the public from airborne radionuclides as a result of normal operations and Category 1 and Category 2 event sequences.

Twelve base cases (Table 18, Runs 1 to 12) and four sensitivity cases (Table 18, Runs 13 and 16) are run using MACCS2 (ORNL 1998) for normal operations, and for Category 1 and Category 2 event sequences involving PWR and BWR SFAs, HLW, and naval SNF. The output file names from the 12 principal runs with four sensitivity runs are listed in Table 18.

Table 18. MACCS2 Computer Runs

Run No.	ATMOS File	Output File	Remarks
1	LASUR01.INP	LASUR01.OUT	Surface releases from 1 percent of an annual throughput of 3000 MTHM, average PWR inventory with HEPA filters, elevated release
2	LASUR02.INP	LASUR02.OUT	Oxidation of 154 damaged fuel rods per year, average PWR inventory with HEPA filters, elevated release
3	LASUB01.INP	LASUB01.OUT	Subsurface releases from waste package surface contamination, neutron-activated air and dust without HEPA filters, ground-level release
4	LAPWR01.INP	LAPWR01.OUT	1 assembly, average PWR inventory with HEPA filters, ground-level release
5	LAPWR02.INP	LAPWR02.OUT	1 assembly, maximum PWR inventory without HEPA filters, ground-level release
6	LABWR01.INP	LABWR01.OUT	1 assembly, average BWR inventory with HEPA filters, ground-level release
7	LABWR02.INP	LABWR02.OUT	1 assembly, maximum BWR inventory without HEPA filters, ground-level release
8	LAHLW01.INP	LAHLW01.OUT	1 SRS HLW canister inventory without HEPA filters, ground-level release
9	LAHLW02.INP	LAHLW02.OUT	1 Hanford Site HLW canister inventory without HEPA filters, ground-level release
10	LAHLW03.INP	LAHLW03.OUT	1 West Valley HLW canister inventory without HEPA filters, ground-level release
11	LAHLW04.INP	LAHLW04.OUT	1 INEEL HLW canister inventory without HEPA filters, ground-level release
12	LANAV01.INP	LANAV01.OUT	1 naval SNF canister inventory without HEPA filters, ground-level release
13	LASEN01.INP	LASEN01.OUT	Sensitivity analysis of Run 1: surface releases from 1 percent of an annual throughput of 3000 MTHM, average PWR inventory with HEPA filters, elevated release. The release duration is changed from 24 hr to 1 hr
14	LASEN02.INP	LASEN02.OUT	Sensitivity analysis of Run 4: 1 assembly, average PWR inventory with HEPA filters, ground-level release. The deposition velocity is changed from 0.01 m/s to 0.001 m/s
15	LASEN03.INP	LASEN03.OUT	Sensitivity analysis of Run 4: 1 assembly, maximum PWR inventory with HEPA filters, ground-level release
16	LASEN04.INP	LASEN04.OUT	Sensitivity analysis of Run 4: 1 assembly, average PWR inventory without HEPA filters, ground-level release

BWR = boiling water reactor; HEPA = high-efficiency particulate air; HLW = high-level radioactive waste; INEEL = Idaho National Engineering and Environmental Laboratory; PWR = pressurized water reactor; SNF = spent nuclear fuel; SRS = Savannah River Site.

Source: ORNL 1998

The ATMOS file in MACCS2 provides the radiological inventory, defines the radial distances of 16 sector wind rose, establishes the dry or wet deposition factors, provides the release fractions for the predefined isotopic groups, and integrates the site-specific weather data with a selected frequency distribution. Only the ATMOS file is changed when a different case is run. The input parameters used in ATMOS, EARLY, and CHRONC output files for each of the sixteen cases are given in Attachment C.

Normal operations and Category 1 event sequence dose results are from the MACCS2 (ORNL 1998) output file at either the mean or 50th percentile probability level, depending on which probability level gives a higher dose.

Category 2 event sequence dose results are from the MACCS2 (ORNL 1998) output file at the 95th percentile probability level. Calculated doses are converted from Sv to mrem using a conversion factor of 1 Sv equal to 1.0×10^5 mrem. The majority of the dose is received during the initial plume passage.

Public dose includes contributions from inhalation and cloudshine during plume passage plus resuspension inhalation, ingestion, and groundshine pathways during the long-term exposure period. Internal doses are calculated using a dose commitment period of 50 years.

The values of DR, ARF, RF, $(LPF)_{cask}$, and $(LPF)_{fac}$ used in these MACCS2 (ORNL 1998) computer runs are summarized in Table 19.

Table 19. Values of DR, ARF, RF, $(LPF)_{cask}$, $(LPF)_{fac}$, and $(LPF)_{HEPA}$ Used in MACCS2

Run No.	DR	ARF	RF	$(LPF)_{cask}$	$(LPF)_{fac}$	$(LPF)_{HEPA}$
1	0.01[a]	0.3 (g) 2.0E-04 (c) 3.0E-05 (p) 1.5E-02 (crud)[b]	1.0	1.0	1.0	1.0(g) 1.0E-04(c) 1.0E-04(p) 1.0E-04(crud)
2	1.0	0.3 (g) 2.0E-04 (c) 1.2E-04 (p) 1.5E-02 (crud)[b]	1.0	1.0	1.0	1.0(g) 1.0E-04(c) 1.0E-04(p) 1.0E-04(crud)
3	1.0	1.0	1.0	1.0	1.0	1.0
4	1.0	0.3 (g) 2.0E-04 (c) 3.0E-05 (p) 1.5E-02 (crud)[b]	1.0	1.0	1.0	1.0(g) 1.0E-04(c) 1.0E-04(p) 1.0E-04(crud)
5	1.0	0.3 (g) 2.0E-04 (c) 5.9E-07 (p) 1.5E-02 (crud)[b]	1.0	1.0 (g) 0.1 (c) 0.1 (p) 0.1 (crud)	1.0	1.0
6	1.0	0.3(g) 2.0E-04 (c) 3.0E-05 (p) 1.5E-02 (crud)[b]	1.0	1.0	1.0	1.0(g) 1.0E-04(c) 1.0E-04(p) 1.0E-04(crud)
7	1.0	0.3 (g) 2.0E-04 (c) 5.9E-07 (p) 1.5E-02 (crud)[b]	1.0	1.0 (g) 0.1 (c) 0.1 (p) 0.1 (crud)	1.0	1.0
8	1.0	6.14E-05	1.0	0.01	1.0	1.0
9	1.0	6.14E-05	1.0	0.01	1.0	1.0
10	1.0	6.14E-05	1.0	0.01	1.0	1.0
11	1.0	6.14E-05	1.0	0.01	1.0	1.0

Table 19. Values of DR, ARF, RF, $(LPF)_{cask}$, $(LPF)_{fac}$, and $(LPF)_{HEPA}$ Used in MACCS2 (Continued)

Run No.	DR	ARF	RF	$(LPF)_{cask}$	$(LPF)_{fac}$	$(LPF)_{HEPA}$
12	See Note[c]	0.3 (g) 2.0E-04 (c) 3.0E-05 (p) 1.5E-02 (crud)[b]	See Note[d]	1.0	1.0	1.0
13	0.01[a]	0.3 (g) 2.0E-04 (c) 3.0E-05 (p) 1.5E-02 (crud)[b]	1.0	1.0	1.0	1.0(g) 1.0E-04(c) 1.0E-04(p) 1.0E-04(crud)
14	1.0	0.3 (g) 2.0E-04 (c) 3.0E-05 (p) 1.5E-02 (crud)[b]	1.0	1.0	1.0	1.0(g) 1.0E-04(c) 1.0E-04(p) 1.0E-04(crud)
15	1.0	0.3 (g) 2.0E-04 (c) 3.0E-05 (p) 1.5E-02 (crud)[b]	1.0	1.0	1.0	1.0(g) 1.0E-04(c) 1.0E-04(p) 1.0E-04(crud)
16	1.0	0.3 (g) 2.0E-04 (c) 3.0E-05 (p) 1.5E-02 (crud)[b]	1.0	1.0	1.0	1.0

NOTES: (c) = cesium; (g) = fission product gases; (p) = particulate.

[a] DR=0.01 is used in calculating the release fraction input to MACCS2 (ORNL 1998)

[b] For crud, the value shown is the "effective ARF," which is the product of a crud spallation fraction of 0.15 and an ARF of 0.1 (BSC 2004m, Section 6.2.1.3)

[c] Radionuclide releases for Category 2 event sequences involving naval SNF were developed by the Naval Nuclear Propulsion Program (Gisch 2004)

[d] RFs for radionuclide releases for Category 2 event sequences involving naval SNF are identified in Table 5.

ARF = airborne release fraction; DR = damage ratio; LPF = leak path factor; RF = respirable fraction.

Run 1 involves a HEPA-filtered release from an annual throughput of 3,000 MTHM of PWR SFAs in a DTF, or the FHF, while Run 2 involves a HEPA-filtered release from oxidation of 154 damaged fuel rods per year in a DTF or the FHF.

Run 3 simulates an unfiltered subsurface release of waste package surface contamination, neutron-activated air, and silica dust during normal operations.

Run 4 involves a HEPA-filtered release from a drop or collision of an Average PWR assembly while Run 5 simulates an unfiltered release from a drop or collision of a Maximum PWR assembly.

Run 6 involves a HEPA-filtered release from a drop or collision of an Average BWR assembly while Run 6 simulates an unfiltered release from a drop or collision of a Maximum BWR assembly.

Runs 8, 9, 10, and 11 involve unfiltered releases from a drop or collision of an SRS HLW canister, a Hanford Site HLW canister, a West Valley HLW canister, and an INEEL HLW canister.

Run 12 simulates an unfiltered release from a drop or collision of a naval SNF canister.

Run 13 is a variation of Run 1; the radionuclide release duration input parameter used in Run 1 was changed from 24 hr used in Run 1 to 1 hr used in Run 13.

Run 14 is a variation of Run 4; Run 14 changes the particle deposition velocity input parameter used in Run 4 from 0.01 m/s to 0.001 m/s.

Run 15 is a variation of Run 4; Run 15 changes the source term used in Run 4 from one average PWR assembly to one maximum PWR assembly.

Run 16 is a variation of Run 4; Run 4 takes credit for HEPA filters while Run 16 does not.

MACCS2 (ORNL 1998) analyses results are summarized in Tables 17, 18, and 22 to 24.

6.3.1 Doses to Members of the Public from Normal Operations and Category 1 Event Sequences

MACCS2 (ORNL 1998) was used to calculate public TEDE to members of the public onsite at 100 m and at the site boundary of 11 km from normal operations (Table 20) and Category 1 event sequences (Table 21).

Table 20. Public Doses from Normal Operations

Releases from:	Public TEDE at 100 m (mrem/year)	Public TEDE at 3 km (mrem/year)	Public TEDE at 11 km (mrem/year)
Normal operations: surface releases from 1 percent of 3,000 MTHM SNF per year and oxidation of 154 damaged commercial SNF rods per year, Average PWR SFAs, HEPA-filtered[a]	3.02E+00	1.94E-01	3.51E-01
Normal operations: subsurface releases of waste package surface contamination, activated air and silica dust, unfiltered[b]	1.17E+01	1.17E-02	1.67E-02

NOTES: [a] Table 18, MACCS2 Run 1 + MACCS2 Run 2
[b] Table 18, MACCS2 Run 3.

HEPA = high-efficiency particulate air; mrem = one thousandth of a rem; MTHM = metric tons of heavy metal; PWR = pressurized water reactor; rem = roentgen equivalent man; TEDE = total effective dose equivalent.

Table 21. HEPA-Filtered Public TEDE for Category 1 Event Sequences

Event Sequence Identifier	Category 1 Events	Location	MAR (No. SFAs)		HEPA-Filtered Public TEDE at 100 m (mrem/event)		HEPA-Filtered Public TEDE at 11 km (mrem/event)	
			PWR[a]	BWR[b]	Avg PWR[a]	Avg BWR[b]	Avg PWR[a]	Avg BWR[b]
GET-03D	Drop of a commercial SNF assembly onto another assembly in a transportation cask	DTF and FHF Waste Transfer Cells	2	2	1.86E+00	5.44E-01	8.64E-03	2.74E-03
GET-03B	Collision involving a commercial SNF assembly	DTF and FHF Waste Transfer Cells	1	1	9.28E-01	2.72E-01	4.32E-03	1.37E-03

NOTES: Avg PWR Fuel: Enrichment = 4.0 percent; Burnup = 48 GWd/MTU; Decay Time = 25 years.
Avg BWR Fuel: Enrichment = 3.5 percent; Burnup = 40 GWd/MTU; Decay Time = 25 years.

[a] Table 18, MACCS2 Run 4
[b] Table 18, MACCS2 Run 6.

Avg = average; BWR = boiling water reactor; DTF = Dry Transfer Facility (DTF 1 or DTF 2); FHF = Fuel Handling Facility; GWd = gigawatt day; HEPA = high-efficiency particulate air; MAR = material at risk; mrem = one thousandth of a rem; MTU = metric tons uranium; PWR = pressurized water reactor; rem = roentgen equivalent man; SFA = spent fuel assembly; SNF = spent nuclear fuel; TEDE = total effective dose equivalent.

The maximally exposed individual at the site boundary of 11 km is assumed to receive doses from the inhalation, resuspension inhalation, air submersion, groundshine, and ingestion pathways for a period of 8,760 hr (Section 4, Assumption 4.20). The onsite individual member of the public, at 100 m or 3 km away from a DTF, the FHF, or a subsurface exhaust shaft, is assumed to receive doses from inhalation, resuspension inhalation, air submersion, and groundshine pathways for a period of 2,000 hr (Section 4, Assumption 4.20). Only HEPA-filtered public TEDEs are reported in Table 21.

In Table 20, the doses at 11 km are larger than the doses at 3 km, because: (1) the doses at 11 km include the ingestion pathway while the doses at 3 km do not, and (2) the doses at 11 km assumes an exposure period of 8,760 hr while the doses at 3 km assume an exposure period of only 2,000 hr.

6.3.2 Sum of Doses from Normal Operations and Category 1 Event Sequences

The dose from an average PWR assembly is higher than the dose from an average BWR assembly (Table 21) and, therefore, the average PWR dose per SFA is used in the calculation of the sum of frequency-weighted Category 1 event sequence doses using:

$$\text{Category 1 doses} = (FA_{drop} \times N_{drop} + FA_{coll} \times N_{coll}) \times D_{pwr} \qquad \text{(Eq. 18)}$$

where,

FA_{drop} = Maximum number of SFAs affected in a drop event
N_{drop} = Number of drops per year

FA_{coll} = Maximum number of SFAs affected in a collision event

N_{coll} = Number of collisions per year

D_{pwr} = Average PWR drop or collision dose per SFA.

The results of the dose aggregation calculation using Equation 18 are shown in Table 22.

Table 22. Sum of Public Doses From Normal Operations and Category 1 Event Sequences

Event Sequence Identifier	Releases from:	MAR (1)	Number of events/ year (2)	Public TEDE at 100 m (mrem/ event)[a] (3)	Public TEDE at 11 km (mrem/ event)[b] (4)	Public TEDE at 100 m (mrem/year)[a] (5)=(2)×(3)	Public TEDE at 11 km (mrem/year)[b] (6)=(2)×(4)
NA	Normal operations: surface releases from 64 Average PWR SFAs per year, HEPA-filtered	NA	NA	NA	NA	Annual Dose 3.02E+00[c]	Annual Dose 3.51E-01[c]
NA	Normal operations: subsurface releases of waste package surface contamination, activated air and silica dust, unfiltered	NA	NA	NA	NA	Annual Dose 1.17E-02[c]	Annual Dose 1.67E-02[c]
GET-03D	Drop of a commercial SNF assembly onto one assembly in a transportation cask	2[d]	0.5[e]	1.86E+00[d]	8.64E-03[d]	Frequency Weighted Dose 9.28E-01	Frequency Weighted Dose 4.32E-03
GET-03B	Collision involving a commercial SNF assembly	1[d]	0.5[e]	9.28E-01[d]	4.32E-03[d]	Frequency Weighted Dose 4.64E-01	Frequency Weighted Dose 2.16E-03
					Total:	4.43E+00	3.74E-01

NOTES: [a] Onsite member of the public is located at 100 m away from a DTF or the FHF and at 3 km away from the subsurface exhaust shafts and is exposed to radiation for a period of 2,000 hr (Section 4, Assumption 4.20)

[b] Offsite member of the public is located at 11 km away from a DTF or the FHF, and at 8 km away from the subsurface exhaust shafts and is exposed to radiation for a period of 8,760 hr (Section 4, Assumption 4.20)

[c] Table 20, columns 1, 2, and 3

[d] Table 21

[e] Table 9, column 3.

DTF = Dry Transfer Facility (DTF 1 or DTF 2); FHF = Fuel Handling Facility; HEPA = high-efficiency particulate air [filter]; NA = not applicable; PWR = pressurized water reactor; SNF = spent nuclear fuel.

Three combinations of Category 1 event sequences were identified for consideration in the consequence analysis (BSC 2004k, Section 7.2.1):

- Two commercial SNF assembly drops within 1 year
- Two commercial SNF assembly collisions within 1 year
- One drop and one collision of commercial SNF assemblies within 1 year.

The collision event involves one PWR or BWR SFA; the drop event potentially affects two PWR or BWR SFAs.

The case of the two commercial SNF assembly drops has the highest MAR (i.e., four SFAs) and, hence, has the highest radiation dose among the three cases.

The radiation doses resulting from two commercial SNF assembly drops and normal operations are 6.64E+00 for public exposure at 100 m and 3.77E-01 mrem/year for public exposure at the site boundary.

6.3.3 Doses to Members of the Public from Category 2 Event Sequences

The outputs from MACCS2 (ORNL 1998) dose calculations for members of the public from Category 2 event sequences are inhalation, resuspension inhalation, ingestion, cloudshine, and groundshine doses at 11 km (Section 4, Assumption 4.15). Table 10 shows that the bounding Category 2 event sequences are:

- Event GET-01A(a) for a drop and breach of 36 PWR SFAs
- Event GET-01A(b) for a drop and breach of 74 BWR SFAs
- Event GET-02B(n) for a drop and breach of one naval SNF canister
- Event GET-02B(a) for a drop and breach of five HLW canisters from SRS
- Event GET-02B(b) for a drop and breach of five HLW canisters from the Hanford Site
- Event GET-02B(c) for a drop and breach of five HLW canisters from West Valley
- Event GET-02B(d) for a drop and breach of five HLW canisters from INEEL.

Doses to members of the public from these bounding Category 2 event sequences are summarized in Tables 23 and 24.

In Table 23, the bounding Category 2 public doses are shown. Public doses from the drop of an SNF canister, an HLW canister, or a DPC are bounded by the doses from dropping a waste package or a transportation cask containing Maximum PWR or Maximum BWR SFAs.

The results of public dose calculations for HLW and naval SNF show that breached SRS HLW canisters have the highest dose consequences (Table 24).

Table 23. Bounding Unfiltered Public Doses for Category 2 Event Sequences

Bounding Category 2 Events	TEDE (mrem/event)	Highest TODE[a] (mrem/event)	SDE (mrem/event)	LDE (mrem/event)
Event GET-01A(a): Drop and breach of a transportation cask containing 36 maximum PWR SFAs without HEPA filters[b]	4.76E+01	1.47E+02	1.95E+01	6.71E+01
Event GET-01A(b): Drop and breach of a BWR transportation cask containing 74 Maximum BWR SFAs without HEPA filters[c]	3.71E+01	9.78E+01	1.40E+01	5.11E+01

NOTES: Maximum PWR Fuel: enrichment = 5.0 percent; burnup = 80 GWd/MTU; decay time = 5 years.
Maximum BWR Fuel: enrichment = 5.0 percent; burnup = 75 GWd/MTU; decay time = 5 years.

[a] The bone surface dose is the highest TODE
[b] Table 18, MACCS2 Run 5
[c] Table 18, MACCS2 Run 7.

BWR = boiling water reactor; CDE = committed dose equivalent; DDE = deep dose equivalent; HEPA = high-efficiency particulate air; LDE = lens dose equivalent; mrem = one thousandth of a rem; PWR = pressurized water reactor; rem = roentgen equivalent man; SDE = shallow dose equivalent to skin; TODE = total organ dose equivalent.

Table 24. Unfiltered Public Doses From Breached Vitrified High-level Radioactive Waste Canisters and Naval Spent Nuclear Fuel Canisters

Bounding Category 2 Events	TEDE (mrem/event)	Highest TODE[a] (mrem/event)	SDE (mrem/event)	LDE (mrem/event)
Event GET-02B(n): Drop and breach of one naval SNF canister without HEPA filters[b]	1.13E+00	5.86E+00	4.91E-02	1.18E+00
Event GET-02B(a): Drop and breach of five SRS HLW canisters without HEPA filters[c]	3.85E+00	4.18E+01	8.50E-03	3.86E+00
Event GET-02B(b): Drop and breach of five Hanford Site HLW canisters without HEPA filters[d]	4.79E-01	6.22E+00	8.30E-03	4.87E-01
Event GET-02B(c): Drop and breach of five West Valley HLW canisters without HEPA filters[e]	4.68E-01	6.83E+00	3.58E-03	4.72E-01
Event GET-02B(d): Drop and breach of five INEEL HLW canisters without HEPA filters[f]	7.07E-02	5.23E-01	1.85E-03	7.26E-02

NOTES: [a] The bone surface dose is the highest TODE for HLW canisters and naval SNF canisters
[b] Table 18, MACCS2 Run 12
[c] Table 18, MACCS2 Run 8
[d] Table 18, MACCS2 Run 9
[e] Table 18, MACCS2 Run 10
[f] Table 18, MACCS2 Run 11.

HEPA = high-efficiency particulate air; HLW = high-level radioactive waste; INEEL = Idaho National Engineering and Environmental Laboratory; LDE = lens dose equivalent; mrem = one thousandth of a rem; rem = roentgen equivalent man; SDE = shallow dose equivalent to skin; SRS = Savannah River Site; TEDE = total effective dose equivalent; TODE = total organ dose equivalent.

6.3.4 Sensitivity Analysis and Treatment of Uncertainty

This section discusses sensitivity analysis performed for and treatment of uncertainties in public doses. A discussion of uncertainties in worker doses is beyond the scope of this report because the calculated worker doses presented in this calculation were provided by various references.

6.3.4.1 Sensitivity Analysis

A maximum release duration of 24 hr allowed in MACCS2 (ORNL 1998) is used for normal operations (Section 4, Assumption 4.6). An initial investigation (Table 18, Run 13) was performed whereby doses from a 1-hr and a 24-hr release are calculated; doses from the shorter 1-hr release duration are higher than the longer 24-hr release duration.

The finding of the investigation is considered reasonable because the radionuclides inhaled and ingested have a slightly longer time to decay and produce alpha and beta particles. Therefore, the use of a release duration of 1 day instead of 1 year with the same amount of radionuclide releases for normal operations is conservative.

For MACCS2 (ORNL 1998) base cases, a particle deposition velocity of 0.01 m/s is used. For sensitivity analysis, a particle deposition velocity of 0.001 m/s is used in MACCS2 Run 14 (Table 18) with results showing a dose decrease by a factor of about 4.

MACCS2 Run 15 (Table 18) has shown that the calculated doses increase by approximately a factor of 3 by use of a maximum PWR assembly instead of an average PWR assembly.

MACCS2 Run 16 (Table 18) has shown that without HEPA filters the calculated doses at 100 m and 11 km away from the release point for a drop event involving two PWR SFAs could exceed the 100 mrem and 15 mrem dose limits for onsite and offsite members of the public, respectively.

6.3.4.2 Treatment of Uncertainty

Uncertainties in consequence analyses are addressed by using conservative inputs and assumptions. For example, for calculating public and worker doses for normal operations, it is conservatively assumed that 1 percent of received SFAs contain fuel rods with damaged cladding. As stated in Section 6.1.2.1, historical fuel discharge data have shown that the fuel rod failure rate is about 0.01 percent, versus 0.02 to 0.07 percent in the first 20 years of commercial nuclear power (BSC 2003c, p. G-6). This shows that the true fuel rod failure rate is at least one order of magnitude smaller than the 1 percent failure rate used in the consequence analyses. For calculating public and worker doses for Category 1 event sequences involving drops and collisions, it is conservatively assumed that 100 percent fuel rod cladding is breached. This is at least a factor of five larger than the best estimate value.

A comparison between MACCS2 Run 4 and Run 15 (Table 18) has shown that using the maximum PWR assembly inventory and the 95th percentile χ/Q value instead of the average PWR assembly inventory and the mean χ/Q value, the calculated doses for Category 1 event sequences involving drops and collisions increase by a factor of about 9. Even with this increase, the sum of normal operational doses and frequency-weighted Category 1 event sequence doses is still well within the regulatory dose limits given in Section 7.

Uncertainties in calculating public doses for Category 2 event sequences are addressed by using conservative inputs and bounding assumptions. The bounding assumptions include the use of the bounding MAR (i.e., the maximum possible number of SFAs allowed in a certified cask) for a transportation cask, bounding source terms, 100 percent fuel rod cladding breach, and 95th percentile weather condition (i.e., χ/Q values). The bounding MAR (i.e., 36 PWR or 74 BWR SFAs in a transportation cask) is at least a factor of 1.5 higher than the average value of the MAR. A comparison between MACCS2 Run 4 and Run 15 (Table 18) has shown that the bounding PWR and BWR source terms are at least a factor of 3 higher than the average PWR and BWR source terms.

The assumption of a 100 percent fuel rod cladding breach (Section 4, Assumption 4.10) is very conservative in view of the fact that historical SNF drop incidents at nuclear power plants showed only minimum damage to the cladding. Run 4 (Table 18) has shown that the use of the 95th percentile weather condition is about a factor of 3 higher than the mean value.

The information on conservatism provided in this section ensures that uncertainties in inputs are treated with a reasonable degree of conservatism.

6.3.5 Direct Radiation Exposures During Normal Operations, Category 1 Event Sequences, and Category 2 Event Sequences

Offsite direct exposures originate only from the surface facilities, because subsurface facilities, such as the emplacement drifts, are shielded by massive rock. Surface facilities with potential sources of contribution to offsite direct exposures include the transportation cask buffer/staging area and the SNF aging system.

Other facilities, such as the CHF, DTF 1, DTF 2, and the FHF, provide concrete shielding for the exterior walls except for the entrance vestibules. This shielding typically reduces the dose rate outside the building to 0.25 mrem/hr (BSC 2004n, Section 4.9.1.3) except during cask handling operations around the vestibule. These dose rates result in a negligible contribution to the public dose at the site boundary located at 11 km (Section 4, Assumption 4.15) from the surface facilities.

The transportation cask buffer/staging area provides space for staging of shielded truck and rail casks. The dose rate from an individual cask is regulated by the NRC (e.g., 10 mrem/hr at any point 2 m from the outer lateral surface of the transport vehicle specified in 10 CFR 71.47(b)(3) for exclusive use shipment only). With multiple casks in the staging area, the dose rate will increase, depending on the cask staging configurations. This radiation level affects the worker dose, but poses no impact on the public dose at the site boundary because of the large distance factor.

In addition, 10 CFR 20.1301(a)(2) requires that each licensee shall conduct operations so that the dose in any unrestricted area from external sources does not exceed 0.002 rem (2 mrem) in any 1 hr. This requirement applies to normal operations and Category 1 event sequences.

The dose requirements in 10 CFR Part 20 include contributions from direct exposures. Direct exposures from the SNF aging system arise from the effects of neutron and gamma skyshine. These effects depend on site-specific fuel characteristics of initial enrichment, burnup, and cooling time, along with facility layout, site layout, and site boundary characteristics.

Skyshine radiation dose calculations are typically performed with the MCNP code (Briesmeister 1997). Aging cask vendors have included the skyshine analysis in their safety analysis reports submitted to the NRC to demonstrate regulatory compliance, such as *Topical Safety Analysis Report for the HI-Storm 100 Cask System, HOLTEC Report HI-951312, Revision 2, NRC Docket No. 72-1014* (NRC 1997, Section 5.4.3, p. 5.4-4).

NRC (1997, Section 5.4.3, p. 5.4-4) shows that the skyshine dose at 200 m from an independent spent fuel storage facility is in the range of 11.6 to 19.3 mrem/year (NRC 1997, Table 5.1.9, p. 5.1-15) for array configurations 2 × 3, 2 × 4, and 2 × 5. This dose would be further reduced by permanent or temporary shielding. With this dose level, the public dose at the site boundary from skyshine radiation is negligible, based on the distance factor and attenuation by air.

6.3.6 Doses to Members of the Public Being ALARA

Regulation 10 CFR 20.1101(b) states that the licensee shall use to the extent practical both procedures and engineering controls based upon sound radiation protection principles to achieve occupational doses and doses to members of the public that are ALARA.

To implement the ALARA goals of 10 CFR 20.1101(b), the calculated offsite public TEDE is compared with the 10 mrem/year offsite ALARA dose constraint (Section 7) along with other project ALARA goals.

6.4 WORKER DOSE RESULTS

The results of worker dose calculations are discussed in this section.

6.4.1 Normal Operations

6.4.1.1 Worker Dose From Airborne Releases

The annual doses to surface facility workers from potential surface facility normal operational releases are calculated and summarized in BSC (2004g, Table 13) and BSC (2004e, Table 5) for a full time worker who works 2,000 hr/year (BSC 2004g, Section 4.3.10).

Results show that workers in a DTF, among the 30 receptor locations, receive the highest TEDE (Table 25). The estimated maximum annual surface facility worker dose from airborne releases is 1.6E+00 mrem/year TEDE (Table 25).

Table 25. Annual Surface Facility Worker Doses From Normal Operations Releases

Released from	TEDE[a] (mrem/year)	Highest TODE[b] (mrem/year)	SDE[b] (mrem/year)	LDE[b] (mrem/year)
DTF 1	8.4E-01	2.9E+00	3.5E+00	4.3E+00
DTF 2	4.6E-01	1.6E+00	1.9E+00	2.4E+00
FHF	2.7E-01	1.5E+00	1.9E+00	2.3E+00
Maximum of 17A and 17B	2.2E-03	NC	NC	NC
Subsurface Exhaust Shafts	1.2E-02	NC	NC	NC
Total[c]	1.6E+00	6.0E+00	7.3E+00	9.0E+00

NOTES: [a] BSC 2004e, Table 5
[b] BSC 2004g, Table 13
[c] Total is the sum of DTF 1, DTF 2, FHF, Max. of 17A and 17B, and subsurface exhaust shafts.

DTF = Dry Transfer Facility; FHF = Fuel Handling Facility; LDE = lens dose equivalent; mrem = one thousandth of a rem; NC = not calculated; rem = roentgen equivalent man; SDE = shallow dose equivalent to skin; SNF = spent nuclear fuel; TEDE = total effective dose equivalent; TODE = total organ dose equivalent.

Three potential sources of airborne radioactive material releases were calculated for the subsurface repository under normal operations, which are (BSC 2004g, Section 6):

- Radioactive contamination suspended from the external surfaces of the emplaced waste packages
- Neutron activation products generated from irradiation of ventilation air inside the emplacement drifts
- Neutron activation products generated from irradiation of the host rocks inside the emplacement drifts.

Annual TEDEs to subsurface workers from potential normal operational releases are calculated and summarized in BSC (2004e, Table 5). The estimated maximum annual subsurface facility worker dose from airborne releases is 1.8E-01 mrem/year TEDE (Table 26).

The annual maximum dose to a repository worker was calculated for a full time worker who works 2,000 hr/year at the access main that is taking air from Intake Shaft 1, about 410 m from Exhaust Shaft 1 (BSC 2004g, Section 4.4.6). Airborne radioactive contamination in the access main may result from re-entry of the released material from the exhaust shafts.

Table 26. Annual Subsurface Facility Worker Doses From Normal Operations Releases

Released from	TEDE (mrem/year)	Highest TODE (mrem/year	SDE (mrem/year)	LDE (mrem/year)
DTF 1	2.9E-02	NC	NC	NC
DTF 2	2.0E-02	NC	NC	NC
FHF	3.5E-02	NC	NC	NC
Maximum of 17A and 17B	1.8E-03	NC	NC	NC
Subsurface Exhaust Shafts	9.1E-02	NC	NC	NC
Total	1.8E-01	NC	NC	NC

DTF = Dry Transfer Facility; FHF = Fuel Handling Facility; LDE = lens dose equivalent; mrem = one thousandth of a rem; NC = not calculated; rem = roentgen equivalent man; SDE = shallow dose equivalent to skin; TEDE = total effective dose equivalent; TODE = total organ dose equivalent.

Source: BSC 2004e, Table 5

6.4.1.2 Worker Dose From Direct Radiation

Annual worker doses from direct radiation from contained sources, such as transportation casks and waste packages, during normal operations were calculated for the CHF, DTF 1, DTF 2, the FHF, the TCRRF, the Remediation Facility, the SNF aging system, and the subsurface facility (Table 27).

The highest annual dose to subsurface facility workers from exposure to direct radiation is 380 mrem/year for an emplacement/retrieval locomotive operator (Table 27), which does not exceed the ALARA goal to minimize the number of individuals that have the potential of receiving more than 500 mrem/year TEDE (BSC 2004n, Section 4.9.3.3).

The highest annual dose to surface facility workers from exposure to direct radiation is 2,200 mrem/year for a cask handling/emplacement/retrieval worker in the SNF aging system (Table 27). This dose rate exceeds the ALARA goal to minimize the number of individuals that have the potential of receiving more than 500 mrem/year TEDE (BSC 2004n, Section 4.9.3.3). For the final design, ALARA design considerations will be included to achieve the ALARA goal.

Table 27. Annual Worker Doses From Direct Radiation From Facility Normal Operations

Facility	Worker Group	Individual Worker Dose (mrem/year)[a]
CHF	Cask/Waste Receipt Operator	660[b]
	Health Physics Technician	420[b]
	Transfer Bay Operator	100[b]
DTF	Cask/Waste Receipt Operator	1,840[c]
	Health Physics Technician	800[c]
	Transfer Bay Operator	100[c]
	Electrical, Mechanical, HVAC, Utility Maintenance	26[c]
	General Facility Maintenance and Radiation Monitoring	100[c]
FHF	Cask & Waste Receipt Operator	770[d]
	Health Physics Technician	260[d]
	Transfer Bay Operator	100[d]
Remediation	Remediation Operator	170[e]
	Electrical, Mechanical, HVAC, Utility Maintenance	6[e]
	General Facility Maintenance and Radiation Monitoring	18[e]
SNF Aging System	Cask Handling/Emplacement/Retrieval	2,200[f]
	Health Physics Technician	820[f]
	Cask Inspection	300[f]
	Cask Maintenance	280[f]
	Cask Repair	560[f]
Subsurface	Emplacement/Retrieval Locomotive Operator	380[g]
	Ventilation Maintenance	47[g]
	Electrical Maintenance	85[g]
	Mechanical Maintenance	20[g]
	Ground Control Maintenance	3.2[g]
TCRRF	Security Personnel	30[f]
	Transportation Personnel	230[f]
	Cask Operator	620[f]
	Health Physics Technician	110[f]
	Gantry Operator	130[f]

NOTES: [a] Dose values are rounded to two significant values
[b] BSC 2004a, Sections 7.6 and 7.7
[c] BSC 2004c, Table 8, columns 3, 4, and 5;
BSC 2004c,Table 6, column 5 reports the maintenance worker dose
[d] BSC 2004d, Tables 7-1 and 7-2 and Section 7.1.5
[e] BSC 2004h, Table 12, column 2
[f] BSC 2004e, Table 4
[g] BSC 2004i, Table 10, column D.

CHF = Canister Handling Facility; DTF = Dry Transfer Facility (DTF 1 or DTF 2); FHF = Fuel Handling Facility; HVAC = heating, ventilation, and air-conditioning; mrem = one thousandth of a rem; rem = roentgen equivalent man; SNF = spent nuclear facility; TCRRF = Transportation Cask Receipt and Return Facility.

6.4.2 Category 1 Event Sequences

6.4.2.1 Surface Facility Worker Dose

Category 1 event sequence scenarios and estimated maximum worker dose consequences are shown in Table 28. Dose consequences in mrem received by the maximally exposed worker for the Category 1 event sequences are calculated in BSC (2004b, Table 9).

Table 28. Maximum Worker Doses From Category 1 Event Sequences

Facility[a]	Category 1 Event Sequences[a]	χ/Q (s/m^3)[a]	TEDE[a] (mrem)	TODE[a] (mrem)	SDE[a] (mrem)	LDE[a] (mrem)
DTF 1	Drop	1.7E-04	4.6E-01	4.6E+00	5.6E+00	6.1E+00
	Collision[b]	1.7E-04	2.3E-01	2.3E+00	2.8E+00	3.0E+00
DTF 2	Drop	1.8E-05	4.9E-02	4.9E-01	6.0E-01	6.5E-01
	Collision[b]	1.8E-05	2.5E-02	2.5E-01	3.0E-01	3.2E-01
FHF	Drop	2.4E-04	6.6E-01	6.6E+00	7.9E+00	8.6E+00
	Collision[b]	2.4E-04	3.3E-01	3.3E+00	4.0E+00	4.3E+00

NOTES: [a] BSC 2004b, Table 9
[b] Collision doses are 1/2 of the SFA drop doses.

DTF = Dry Transfer Facility; FHF = Fuel Handling Facility; LDE = lens dose equivalent; mrem = one thousandth of a rem; rem = roentgen equivalent man; SDE = shallow dose equivalent to skin; TEDE = total effective dose equivalent; TODE = total organ dose equivalent.

6.4.2.2 Subsurface Facility Worker Dose

There are no Category 1 event sequences associated with the subsurface facility. Therefore, there is no subsurface worker dose from Category 1 event sequences.

6.4.3 Sum of Worker Doses From Normal Operations and Category 1 Event Sequences

Calculated worker doses do not exceed the Category 1 dose limits (Table 1) for releases from normal operations and Category 1 event sequences.

Aggregation of worker doses calculated for normal operations and Category 1 event sequences (Table 28) is shown in Table 29.

The combination of Category 1 event sequences is the same as that discussed in Section 6.3.2.

Table 29. Sum of Worker Doses From Normal Operations and Category 1 Event Sequences

Event Sequence Identifier	Releases from:	FA_{drop} or FA_{coll} (1)	N_{drop} or N_{coll} (2)	D_{pwr} (mrem/ event) (3)	Frequency-Weighted Dose (4)=(2)×(3) or Annual Dose (mrem/year)
NA	Normal operations: surface and subsurface airborne releases, HEPA-filtered	NA	NA	NA	1.6[a]
NA	Normal operations: direct radiation dose to a cask and waste Receipt operator at a DTF	NA	NA	NA	2,200[b]
GET-03D	Drop of a commercial SNF assembly onto one assembly in a transportation cask	2	0.5	0.66[c]	0.33
GET-03B	Collision involving a commercial SNF assembly	1	0.5	0.33[c]	0.17
				Total:	2,202

NOTES: [a] Table 25, column 2
[b] Table 27, column 3
[c] Table 28, column 4.

DTF = Dry Transfer Facility (DTF 1 or DTF 2); HEPA = high-efficiency particulate air; mrem = one thousandth of a rem; NA = not applicable; rem = roentgen equivalent man; SNF = spent nuclear fuel.

The case of two commercial SNF assembly drops has the highest MAR (i.e., 4 SFAs). The radiation dose to the worker resulting from two commercial SNF assembly drops and normal operations is 2.92 rem/year.

7. CONCLUSIONS

MACCS2 (ORNL 1998) and ARCON96 (BSC 2003b) are used to perform the public dose and worker dose calculations. Calculated doses are compared to the performance objectives shown in Table 30.

Calculated public doses and worker doses show that 10 CFR 63.111(a) and 10 CFR 63.111(b) performance objectives are met by taking credit for the HVAC HEPA filters to mitigate the consequences of normal operations and Category 1 event sequences.

The outputs listed in Attachment C have been found reasonable as compared to the inputs, and the results have been found suitable for their intended use.

Table 30. Summary Preclosure Dose Performance Criteria and Evaluation Results

Standard	Limits	Results
Public Exposures[a]		
Preclosure standard: 10 CFR 63.204; Preclosure Performance Objective for normal operations and Category 1 event sequences per 10 CFR 63.111(a)(2)	15 mrem/year[b]	0.4 mrem/year[c] 0.009 mrem/year[d] 0.4 mrem/year[e]
Constraint specified for air emissions of radioactive material to the environment; not a dose limitation per 10 CFR 20.1101(d)[f]	10 mrem/year[b]	0.4 mrem/year[b]
Dose Limits for Individual Members of the Public for normal operations and Category 1 event sequences per 10 CFR 20.1301[f]	100 mrem/year[b, g]	12 mrem/year[b, h]
	2 mrem/hr in any unrestricted area from external sources	< < 2 mrem/hr
Preclosure Performance Objective for any Category 2 event sequence per 10 CFR 63.111 (b)(2)	5 rem[b]	0.05 rem[b]
	50 rem organ or tissue dose other than the lens of the eye	0.2 rem
	15 rem lens of the eye dose	0.07 rem
	50 rem shallow dose to skin	0.02 rem
Workers' Exposures		
Occupational Dose Limits for Adults from normal operational emissions and Category 1 event sequences per 10 CFR 20.1201[i]	5 rem/year[b]	0.001 rem/year[c] 0.001 rem/year[d] 0.001 rem/year[e] 2.2 rem/year[j]
	50 rem/year organ or tissue dose other than the lens of the eye	0.02 rem/year[k]
	15 rem/year lens of the eye dose	0.02 rem/year[l]
	50 rem/year shallow dose to skin	0.03 rem/year[m]

NOTES: [a] Results for public exposures are calculated at the site boundary
[b] TEDE
[c] Sum of TEDEs from normal operations and Category 1 event sequences
[d] EDE from a single Category 1 event sequence in 1 year
[e] TEDE from any combination of Category 1 event sequences in 1 year and from normal operations
[f] 10 CFR 63.111(a)(1)would require operations area to meet the requirements of 10 CFR Part 20
[g] 10 CFR 20.1301(a)(1); dose limit to the extent applicable
[h] Onsite public dose at 100 m (Section 6.3.2)
[i] 10 CFR 63.111(b)(1) would require repository design objectives for Category 1 and normal operations to meet 10 CFR 63.111(a)(1) requirements (10 CFR Part 20)
[j] The highest annual worker dose from direct radiation from facility normal operation
[k] TODE from any combination of Category 1 event sequences in 1 year and from normal operations
[l] Total LDE from any combination of Category 1 event sequences in 1 year and from normal operations
[m] Total SDE from any combination of Category 1 event sequences in 1 year and from normal operations.

EDE = effective dose equivalent; LDE = lens dose equivalent; mrem = one thousandth of a rem; rem = roentgen equivalent man; SDE = shallow dose equivalent to skin; TEDE = total effective dose equivalent; TODE = total organ dose equivalent.

Sources: Tables 18, 22, 23, 28, and 29, and Sections 6.3.2 and 6.4.3.

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8.2 CODES, STANDARDS, AND REGULATIONS

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10 CFR 71. Energy: Packaging and Transportation of Radioactive Material. Readily available.

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8.3 PROCEDURES

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AP-3.13Q, Rev. 3, ICN 3. *Design Control*. Washington, D.C.: U.S. Department of Energy, Office of Civilian Radioactive Waste Management. ACC: DOC.20040202.0006.

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8.4 SOURCE DATA, LISTED BY DATA TRACKING NUMBER

MO0001YMP00001.000Minimum Distances from Selected Yucca Mountain Project Sites to Public Lands. Submittal date: 01/26/2000.

MO0210MXHT8491.000. Mean Seasonal Mixing Height Data (0400 and 1600 PST) for Desert Rock, Nevada from 1984 through 1991. Submittal date: 10/30/02.

MO0306WPMM9802.000. Wind Direction Sector, Wind Speed, Stability Class and Precipitation Data for MACCS2 98-02 Model. Submittal date: 06/06/2003.

MO0304MACCS280.000. Wind Direction Sector, Wind Speed, Stability Class and Precipitation Data for MACCS2 98-00 Model. Submittal date: 04/30/03.

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ATTACHMENT A – SURFACE FACILITY AIRBORNE RELEASE EVALUATION

A.1 Location of Receptors and Releases

The waste handling facilities of the repository consist of multiple buildings constructed near the North Portal to the underground repository. These facilities will receive, handle, stage, prepare, transfer, age, package, seal, transport, and emplace SNF and HLW. The surface facility layout is shown in Figure A-1. The subsurface facility layout is shown in Figure A-2.

The releases from the following waste handling facilities during normal operations are considered:

- Stack of DTF 1 (110)
- Stack of DTF 2 (120)
- Stack of FHF (210)
- SNF Aging System 17A and 17 B
- Subsurface exhaust shafts.

The receptor locations are shown in Figures A-1 and A-2.

A.2 ARCON96-Generated Atmospheric Dispersion Factors

The χ/Q values for airborne releases, at 26 onsite receptor locations, from DTF 1, DTF 2, the FHF, the SNF aging system, and the subsurface exhaust shafts, were calculated using ARCON96 (BSC 2003b) in *GROA Airborne Release Dispersion Factor Calculation* (BSC 2004f, Section 6.3). Receptor and exhaust stack input information is based on drawings (Figures A-1 and A-2). The χ/Q values at receptors for each release source are used to calculate the worker doses for releases from the surface and subsurface facilities from normal operations and Category 1 event sequences. These χ/Q values represent the dispersion factors estimated at the air intake points of the surface and subsurface facilities.

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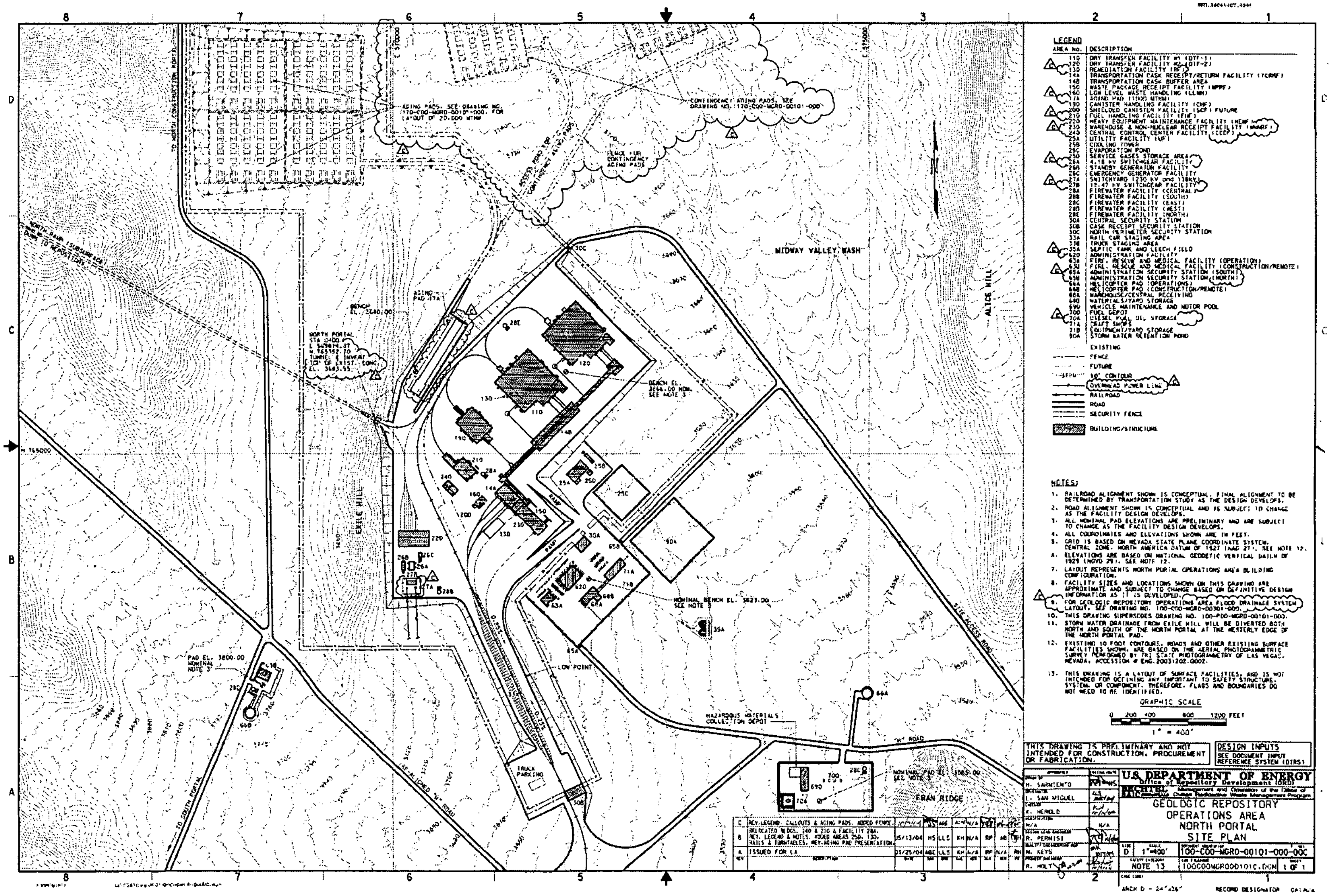

Source: BSC 2004s

NOTE: Notes 1 through 13 are not applicable to this calculation.

Figure A-1. Stack and Intake Location of Repository Surface Facilities

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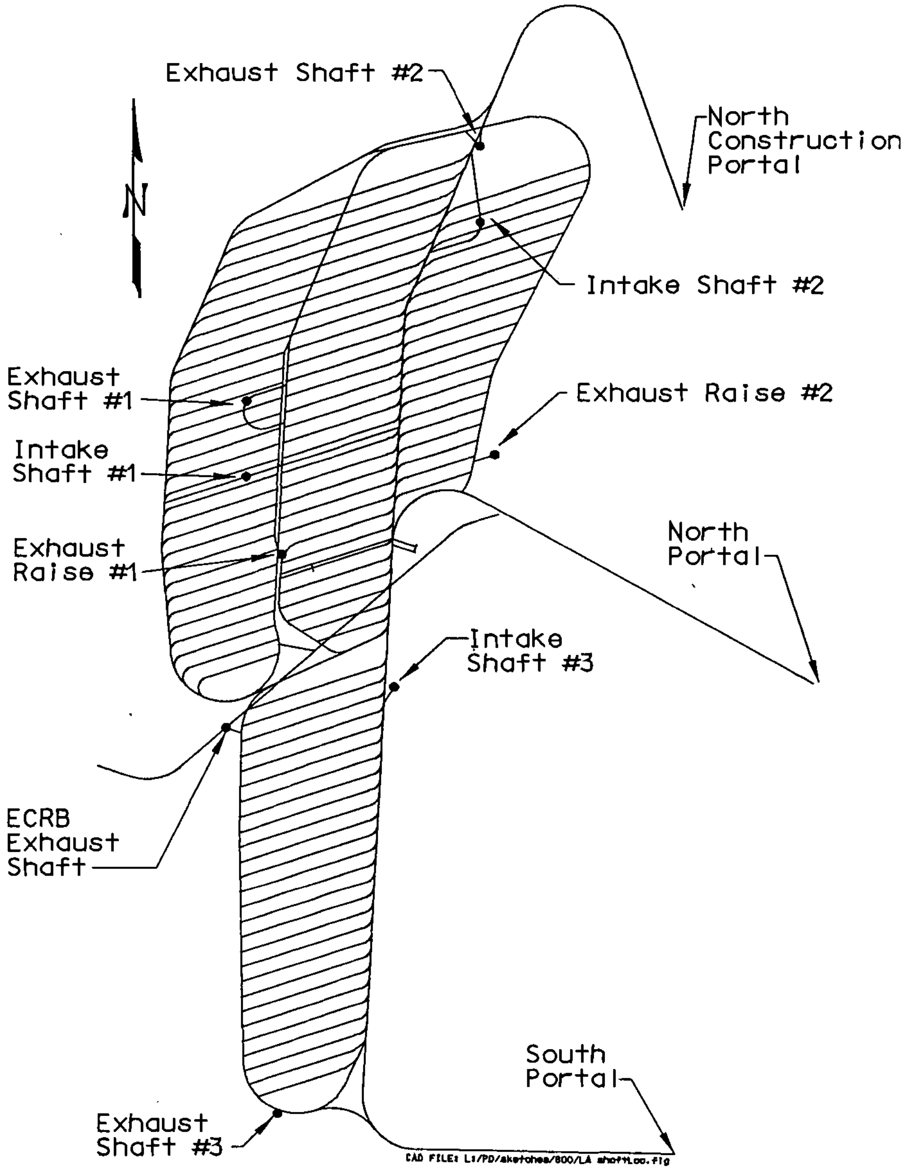


Source: BSC 2003d, Figure 10

Figure A-2. Subsurface Air intake and Exhaust Locations

A.3 MACCS2-Generated Atmospheric Dispersion Factors

The output file for each of the 16 MACCS2 computer runs (Table 18) provides information on the calculated χ/Q values at various distances from the release point (Table A-1). For each distance, the χ/Q values are provided as mean, 50th, 90th, 95th, 99th, and 99.5th percentile values.

The χ/Q values in Table A-1 are used in Section 6.3 to calculate public doses.

Table A-1. χ/Q Values for Normal Operations and Event Sequences

Releases from:	χ/Q Values at Various Distances From Release Point 100 m (s/m^3)	800 m (s/m^3)	8 km (s/m^3)	11 km (s/m^3)
Normal operations in surface facilities: surface releases from 1 percent of 3,000 MTHM SNF per year, Average PWR SFAs, HEPA-filtered[a]	5.62E-05	NC	NC	2.69E-06
Normal operations in subsurface facility: subsurface releases of waste package surface contamination, activated air and silica dust, unfiltered[b]	4.66E-03	1.04E-04	5.26E-06	NC
DTF or FHF: Category 1 event sequences (drops and collisions)[c]	4.94E-04	NC	NC	2.96E-06
CHF, DTF, or FHF: Category 2 event sequences[d]	NC	NC	NC	9.68E-06

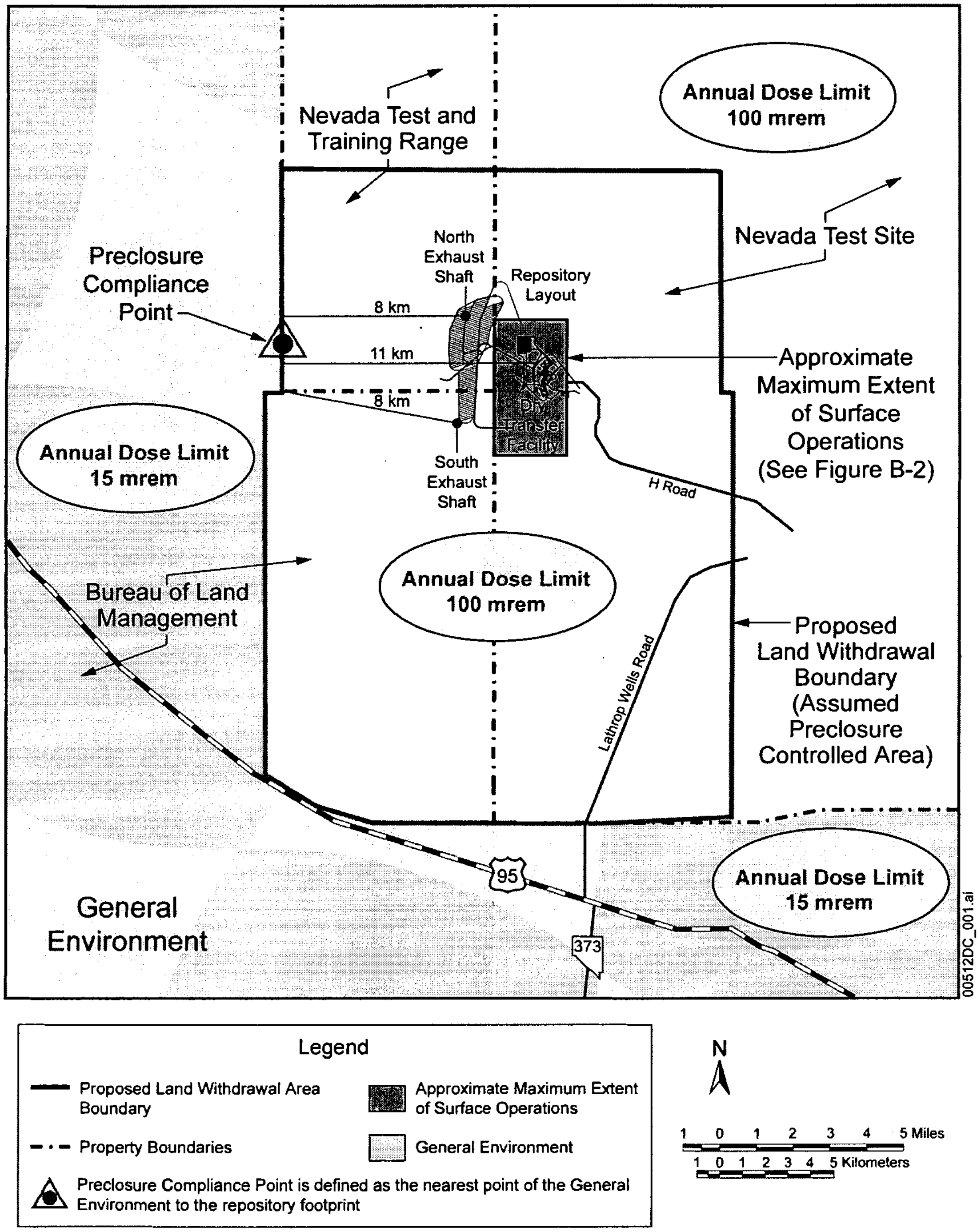
NOTES: [a] Table 18, MACCS2 Run 1
[b] Table 18, MACCS2 Run 3
[c] Table 18, MACCS2 Run 4
[d] Table 18, MACCS2 Run 8.

CHF = Canister Handling Facility; DTF = Dry Transfer Facility (DTF 1 or DTF 2); FHF = Fuel Handling Facility; HEPA = high-efficiency particulate air; MTHM = metric tons of heavy metal; NC = not calculated; PWR = pressurized water reactor; SFA = spent fuel assembly; SNF = spent nuclear fuel.

ATTACHMENT B

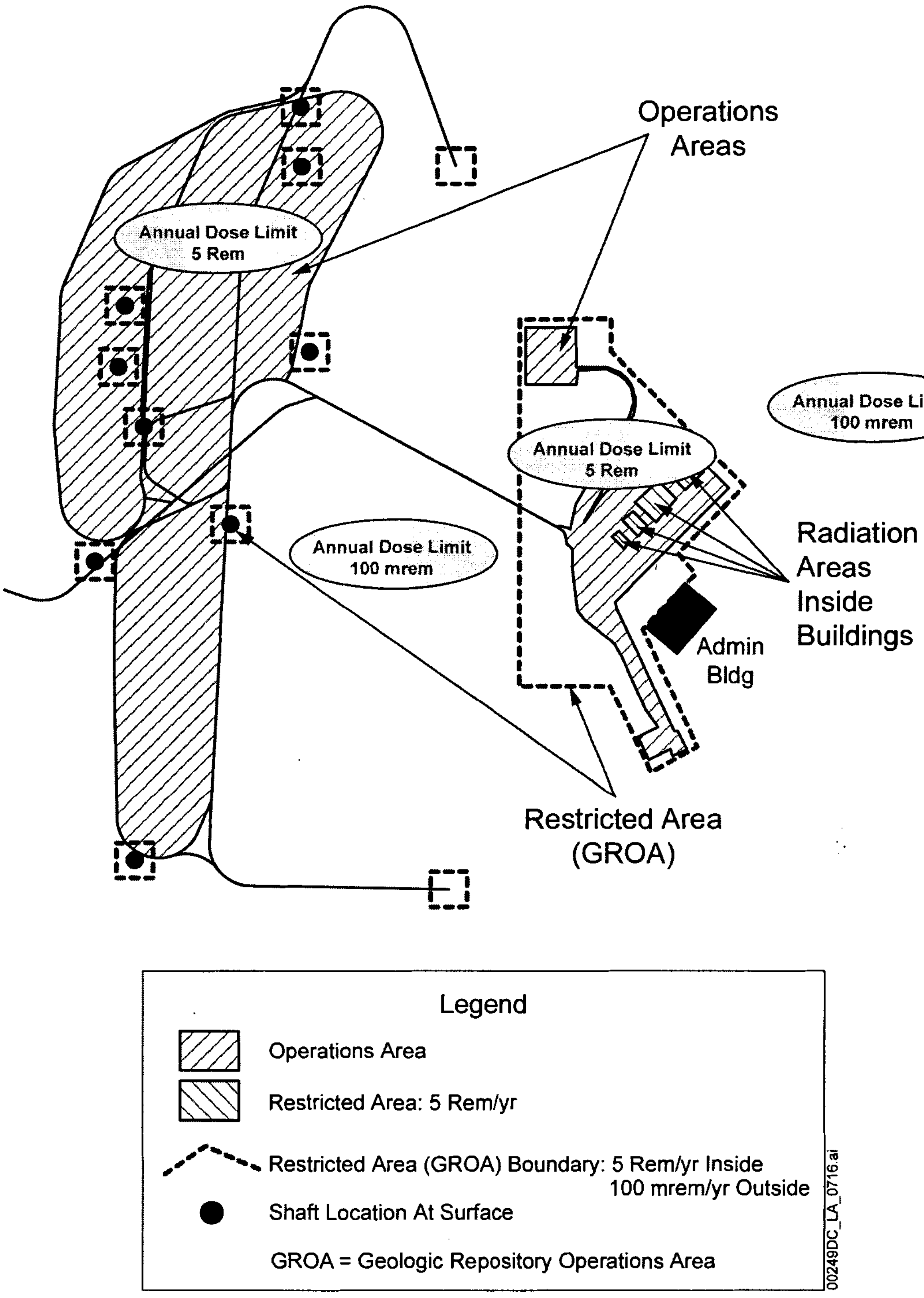
PRECLOSURE RADIATION DOSE LIMITS

Dose limits, both outside and inside the preclosure restricted area, are illustrated in Figures B-1 and B-2. The extent of the geologic repository operations area is to be considered coincident with the restricted area, as illustrated in Figure B-2.



Sources: DTN: MO0001YMP00001.000; Assumption 15 (Section 4); Table 1 (Section 1)

Figure B-1. Minimum Distances From Selected Yucca Mountain Project Sites to Public Lands and Radiation Dose Limits Outside of the Preclosure Controlled Area



Sources: Figures A-1 and A-2 (Attachment A); Table 1 (Section 1)

Figure B-2. Radiation Dose Limits Inside the Preclosure Controlled Area

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ATTACHMENT C – MACCS2 INPUT AND OUTPUT FILE LISTING

MACCS2 (ORNL 1998) input and output files are listed in Table C-1 for sixteen cases (Table 18, Runs 1 to 16) on a compact disk submitted to Document Control.

Table C-1. Listing of MACCS2 Computer Input and Output Files

Run No.	MACCS2 Input and Output File Name	Date	Time	File Size (kB)
1	LASUR01.INP	06/26/2004	1:41 PM	44
1	LASUR01.OUT	06/26/2004	1:58 PM	720
2	LASUR02.INP	10/13/2004	4:03 PM	44
2	LASUR02.OUT	10/13/2004	4:21 PM	720
3	LASUB01.INP	10/27/2004	3:22 PM	44
3	LASUB01.OUT	10/27/2004	3:29 PM	719
4	LAPWR01.INP	06/23/2004	5:53 PM	44
4	LAPWR01.OUT	06/23/2004	6:08 PM	719
5	LAPWR02.INP	06/23/2004	7:35 PM	44
5	LAPWR02.OUT	06/23/2004	7:50 PM	713
6	LABWR01.INP	06/23/2004	8:03 PM	44
6	LABWR01.OUT	06/23/2004	8:17 PM	719
7	LABWR02.INP	06/23/2004	8:25 PM	44
7	LABWR02.OUT	06/23/2004	8:40 PM	713
8	LAHLW01.INP	09/23/2004	1:15 PM	44
8	LAHLW01.OUT	09/23/2004	1:29 PM	710
9	LAHLW02.INP	09/24/2004	11:32 PM	44
9	LAHLW02.OUT	09/24/2004	11:42 PM	711
10	LAHLW03.INP	10/01/2004	1:50 PM	44
10	LAHLW03.OUT	10/01/2004	2:00 PM	710
11	LAHLW04.INP	09/25/2004	9:24 AM	44
11	LAHLW04.OUT	09/25/2004	9:34 AM	709
12	LANAV01.INP	08/30/2004	2:41 PM	46
12	LANAV01.OUT	08/30/2004	2:57 PM	718
13	LASEN01.INP	06/26/2004	2:22 PM	44
13	LASEN01.OUT	06/26/2004	2:39 PM	719
14	LASEN02.INP	06/24/2004	2:51 PM	44
14	LASEN02.OUT	06/24/2004	3:06 PM	719
15	LASEN03.INP	07/13/2004	12:27 PM	44
15	LASEN03.OUT	07/13/2004	12:46 PM	719
16	LASEN04.INP	10/25/2004	3:25 PM	44
16	LASEN04.OUT	10/25/2004	3:45 PM	719
All	YMP1999.INP	07/31/2003	1:27 PM	163
All	YMPCH1.INP	05/17/2004	9:33 AM	11

Table C-1. Listing of MACCS2 Computer Input and Output Files (Continued)

Run No.	MACCS2 Input and Output File Name	Date	Time	File Size (kB)
5, 7, and 8 to 12	YMPEA.INP	08/22/2003	12:43 PM	12
1, 2, 3, 4, 6, 13, and 14	YMPEA1.INP	08/05/2003	3:05 PM	12
1 and 4 to 14	YMPSIT.INP	07/17/2003	7:39 AM	8
3	YSITE8.INP	05/07/2001	7:27 AM	8
All	YUCCALA.BIN	03/10/2004	4:11 PM	1,307
All	YMPD825.INP	02/22/2000	7:38 AM	663